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SECTION 5

GENERATION

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5.1 FOSSIL-FUELED PLANTS

5.1.1 Introduction

America—and much of the world—is becoming increasingly electrified. In 2005, more than half of the electricity generated in the United States came from coal. For the foreseeable future, coal will continue to be the dominant fuel used for electric power production. The low cost and abundance of coal is one of the primary reasons why consumers in the United States benefit from some of the lowest electricity rates of any free-market economy.

The key challenge to keeping coal viable as a generation fuel is to remove the environmental objections to the use of coal in power plants. New technologies are being developed that could virtually eliminate the sulfur, nitrogen, and mercury pollutants released when coal is burned. It may also be possible to capture greenhouse gases that are emitted from coal-fired power plants and prevent them from contributing to global warming concerns.

Research is also underway to increase the fuel efficiency of coal-fueled power plants. Today's plants convert only one-third of coal's energy potential to electricity. New technologies could nearly double efficiency levels in the next 10 to 15 years.

Natural gas is the fastest growing fuel for electricity generation. More than 90% of the power plants to be built in the next 20 years will likely be fueled by natural gas. Natural gas is also likely to be a primary fuel for distributed power generators—mini-power plants that could be sited close to where the electricity is needed.

Natural gas-powered fuel cells are also being developed for future distributed generation applications. Fuel cells use hydrogen that can be extracted from natural gas, or perhaps in the future from biomass or coal.

5.1.2 Thermodynamic Cycles

Rankine Cycle. The cornerstone of the modern steam power plant is a modification of the Carnot cycle proposed by W. J. M. Rankine, a distinguished Scottish engineering professor of thermodynamics and applied mechanics. The temperature-entropy and enthalpy-entropy diagrams of Fig. 5-1 illustrate the state changes for the Rankine cycle. With the exception that compression terminates (state *a*) at boiling pressure rather than the boiling temperature (state *d*), the cycle resembles a Carnot

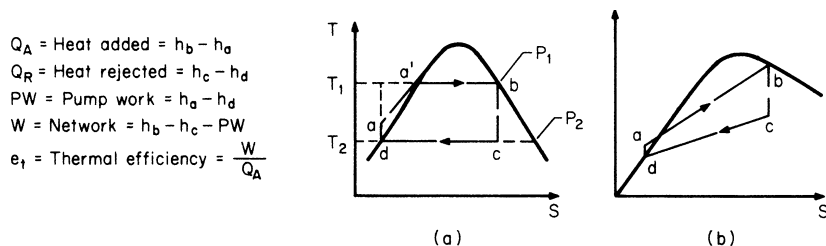


FIGURE 5-1 Simple Rankine cycle (without superheat): (a) temperature-entropy; (b) enthalpy-entropy (Mollier).

cycle. The triangle bounded by $a-d$ and the line connecting to the temperature-entropy curve in Fig. 5-1a signify the loss of cycle work because of the irreversible heating of the liquid from state a to saturated liquid. The lower pressure at state a , compared to $\hat{a}$, makes possible a much smaller work of compression between $d-a$. For operating plants, it amounts to 1% or less of the turbine output.

This modification eliminates the two-phase vapor compression process, reduces compression work to a negligible amount, and makes the Rankine cycle less sensitive than the Carnot cycle to the irreversibilities bound to occur in an actual plant. As a result, when compared with a Carnot cycle operating between the same temperature limits and with realistic component efficiencies, the Rankine cycle has a larger network output per unit mass of fluid circulated, smaller size, and lower cost of equipment. In addition, because of its relative insensitivity to irreversibilities, its operating plant thermal efficiencies will exceed those of the Carnot cycle.

Regenerative Rankine Cycle. Refinements in component design soon brought power plants based on the Rankine cycle to their peak thermal efficiencies, with further increases realized by modifying the basic cycle. This occurred through increasing the temperature of saturated steam supplied to the turbine, by increasing the turbine inlet temperature through constant-pressure superheat, by reducing the sink temperature, and by reheating the working vapor after partial expansion followed by continued expansion to the final sink temperature. In practice, all of these are employed with yet another important modification. The irreversibility associated with the heating of the compressed liquid to saturation by a finite temperature difference is the primary thermodynamic cause of lower thermal efficiency for the Rankine cycle. The regenerative cycle attempts to eliminate this irreversibility by using as heat sources other parts of the cycle with temperatures slightly above that of the compressed liquid being heated.

This procedure of transferring heat from one part of a cycle to another in order to eliminate or reduce external irreversibilities is called “regenerative heating,” which is basic to all regenerative cycles.

The scheme shown in Fig. 5-2 is a practical approach to regeneration. Extraction or “bleeding” of steam at state c for use in the “open” heater avoids excessive cooling of the vapor during turbine expansion; in the heater, liquid from the condenser increases in temperature by ΔT . (Regenerative cycle heaters are called “open” or “closed” depending on whether hot and cold fluids are mixed directly to share energy or kept separate with energy exchange occurring by the use of metal coils.)

The extraction and heating substitute the finite temperature difference ΔT for the infinitesimal dT used in the theoretical regeneration process. This substitution, while failing to realize the full potential of regeneration, halves the temperature difference through which the condensate must be heated in the basic Rankine cycle. Additional extractions and heaters permit a closer approximation to the maximum efficiency of the idealized regenerative cycle, with further improvement over the simple Rankine cycle shown in Fig. 5-1.

Reducing the temperature difference between the liquid entering the boiler and that of the saturated fluid increases the cycle thermal efficiency. The price paid is a decrease in net work produced per pound of vapor entering the turbine and an increase in the size, complexity, and initial cost of the plant. Additional improvements in cycle performance may be realized by continuing to accept the consequences of increasing the number of feedwater heating stages. Balancing cycle thermal efficiency against plant size, complexity, and cost for production of power at minimum cost determines the optimum number of heaters.

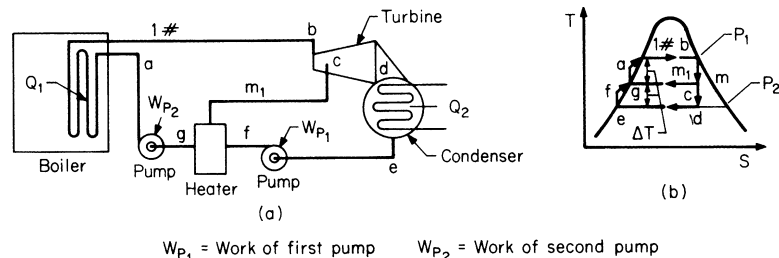


FIGURE 5-2 Single extraction regenerative cycle: (a) flow diagram; (b) temperature-entropy diagram.

Reheat Cycle. The use of *superheat* offers a simple way to improve the thermal efficiency of the basic Rankine cycle and reduce vapor moisture content to acceptable levels in the low-pressure stages of the turbine. But with continued increase of higher temperatures and pressures to achieve better cycle efficiency, in some situations available superheat temperatures are insufficient to prevent excessive moisture from forming in the low-pressure turbine stages.

The solution to this problem is to interrupt the expansion process, remove the vapor for *reheat* at constant pressure, and return it to the turbine for continued expansion to condenser pressure. The thermodynamic cycle using this modification of the Rankine cycle is called the “reheat cycle.” Reheating may be carried out in a section of the boiler supplying primary steam, in a separately fired heat exchanger, or in a steam-to-steam heat exchanger. Most present-day utility units combine superheater and reheater in the same boiler.

Usual central-station practice combines both regenerative and reheat modifications to the basic Rankine cycle. For large installations, reheat makes possible an improvement of approximately 5% in thermal efficiency and substantially reduces the heat rejected to the condenser cooling water. The operating characteristics and economics of modern plants justify the installation of only one stage of reheat except for units operating at supercritical pressure.

Figure 5-3 shows a flow diagram for a 600-MW fossil-fueled reheat cycle designed for initial turbine conditions of 2520-lb/in² (gage) and 1000°F steam. Six feedwater heaters are supplied by exhaust steam from the high-pressure turbine and extraction steam from the intermediate and low-pressure turbines. Except for the deaerating heater (third), all heaters shown are closed heaters. Three pumps are shown: (1) the condensate pump, which pumps the condensate through oil and hydrogen gas coolers, vent condenser, air ejector, first and second heaters, and deaerating heater; (2) the condensate booster pump, which pumps the condensate through fourth and fifth heaters; and (3) the boiler feed pump, which pumps the condensate through the sixth heater to the economizer and boiler. The mass flows noted on the diagram are in pounds per hour at the prescribed conditions for full-load operation.

5.1.3 Reheat Steam Generators

The boiler designer must proportion heat-absorbing and heat-recovery surfaces to make best use of the heat released by the fuel. Waterwalls, superheaters, and reheaters are exposed to convection and radiant heat, whereas convection heat transfer predominates in air heaters and economizers.

The relative amounts of such surfaces vary with the size and operating conditions of the boiler. A small low-pressure heating plant with no heat-recovery equipment has quite a different arrangement from a large high-pressure unit operating on a reheat regenerative cycle and incorporating heat-recovery equipment.

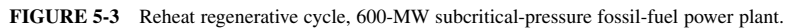
Factors Influencing Boiler Design. In addition to the basics of unit size, steam pressure, and steam temperature, the designer must consider other factors that influence the overall design of the steam generator.

Fuels. Coal, although the most common fuel, is also the most difficult to burn. The ash in coal consists of a number of objectionable chemical elements and compounds. The high percentage of ash that can occur in coal has a serious effect on furnace performance.

At the high temperatures resulting from the burning of fuel in the furnace, fractions of ash can become partially fused and sticky. Depending on the quantity and fusion temperature, the partially fused ash may adhere to surfaces contacted by the ash-containing combustion gases, causing objectionable buildup of slag on or bridging between tubes. Chemicals in the ash may attack materials such as the alloy steel used in superheaters and reheaters.

In addition to the deposits in the high-temperature sections of the unit, the air heater (the coolest part) may be subject to corrosion and plugging of gas passages from sulfur compounds in the fuel acting in combination with moisture present in the flue gas.

Furnace. Heat generated in the combustion process appears as furnace radiation and sensible heat in the products of combustion. Water circulating through tubes that form the furnace wall lining absorbs as much as 50% of this heat, which, in turn, generates steam by the evaporation of part of the circulated water.



Waterwalls usually consist of vertical tubes arranged in tangent or approximately so, connected at top and bottom to headers. These tubes receive their water supply from the boiler drum by means of downcomer tubes connected between the bottom of the drum and the lower headers. The steam,

along with a substantial quantity of water, is discharged from the top of the waterwall tubes into the upper waterwall headers and then passes through riser tubes to the boiler drum. Here the steam is separated from the water, which together with the incoming feedwater is returned to the waterwalls through the downcomers.

Tube diameter and thickness are of concern from the standpoints of circulation and metal temperatures. Thermosyphonic (also called thermal or natural) circulation boilers generally use larger-diameter tubes than positive (pumped) circulation or once-through boilers. This practice is dictated largely by the need for more liberal flow area to provide the lower velocities necessary with the limited head available. The use of small-diameter tubes is an advantage in high-pressure boilers because the lesser tube thicknesses required result in lower outside tube-metal temperatures. Such small-diameter tubes are used in recirculation boilers in which pumps provide an adequate head for circulation and maintain the desired velocities.

Superheaters and Reheaters. The function of a superheater is to raise the boiler steam temperature above the saturated temperature level. As steam enters the superheater in an essentially dry condition, further absorption of heat sensibly increases the steam temperature.

The reheater receives superheated steam which has partly expanded through the turbine. As described earlier, the role of the reheater in the boiler is to re-superheat this steam to a desired temperature.

Superheater and reheater design depends on the specific duty to be performed. For relatively low final outlet temperatures, superheaters solely of the convection type are generally used. For higher final temperatures, surface requirements are larger and, of necessity, superheater elements are located in very high gas-temperature zones. Wide-spaced platens or panels, or wall-type superheaters or reheaters of the radiant type, can then be used. Figure 5-4 shows an arrangement of such platen and panel surfaces. A relatively small number of panels are located on horizontal centers of 5 to 8 ft to permit substantial radiant heat absorption. Platen sections, on 14- to 28-in centers, are placed downstream of the panel elements; such spacing provides high heat absorption by both radiation and convection.

Economizers. Economizers help to improve boiler efficiency by extracting heat from flue gases discharged from the final superheater section of a radiant/reheat unit (or the evaporative bank of an industrial boiler). In the economizer, heat is transferred to the feedwater, which enters at a temperature appreciably lower than that of saturated steam. Generally, economizers are arranged for downward flow of gas and upward flow of water.

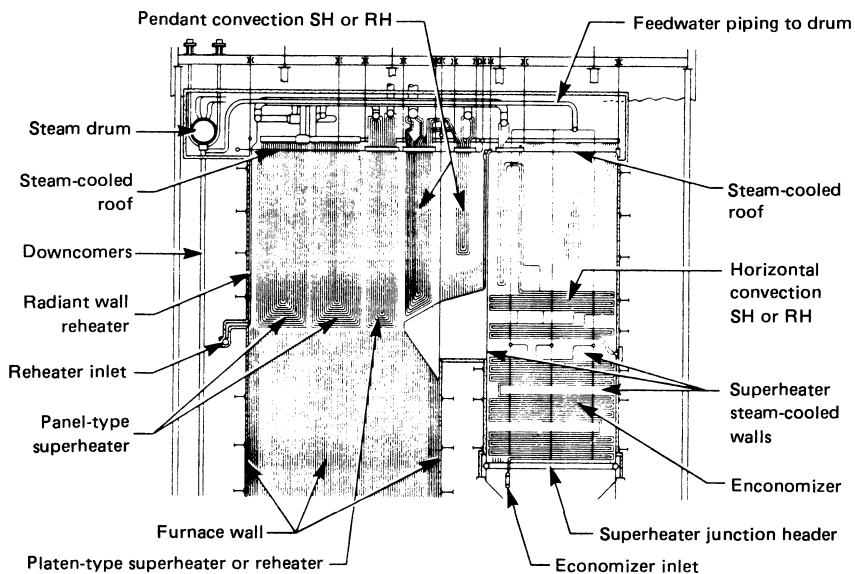


FIGURE 5-4 Arrangement of superheater, reheater, and economizer of a large coal-fired steam generator.

Water enters from a lower header and flows through horizontal tubing constituting the heating surface. Return bends at the ends of the tubing provide continuous tube elements, whose upper ends connect to an outlet header that is in turn connected to the boiler drum by means of tubes or large pipes.

As shown in Fig. 5-4, economizers of a typical utility-type boiler are located in the same pass as the primary or horizontal sections of the superheater, or superheater and reheater, depending on the arrangement of the surface. Tubing forming the heating surface is generally low-carbon steel. Because steel is subject to corrosion in the presence of even extremely low concentrations of oxygen, it is necessary to provide water that is practically 100% oxygen free. In central stations and other large plants, it is a common practice to use deaerators for oxygen removal.

Air Heaters. Steam-generator air heaters have two important and concomitant functions: they cool the gases before they pass to the atmosphere, thereby increasing fuel-firing efficiency; at the same time, they raise the temperature of the incoming air of combustion. Depending on the pressure and temperature cycle, the type of fuel, and the type of boiler involved, one of the two functions will have prime importance.

For instance, in a low-pressure gas- or oil-fired industrial or marine boiler, combustion-gas temperature can be lowered in several ways—by a boiler bank, by an economizer, or by an air heater. Here, an air heater has principally a gas-cooling function, as no preheating is required to burn the oil or gas. If the boiler is a high-pressure reheat unit burning a high-moisture subbituminous or lignitic coal, high preheated-air temperatures are needed to evaporate the moisture in the coal before ignition can take place. Here, the air-heating function becomes primary. Without exception, then, large pulverized-coal boilers either for industry or electric power generation use air heaters to reduce the temperature of the combustion products from the 600 to 800°F level to final exit-gas temperatures of 275 to 350°F. In these units, the combination air is heated from about 80°F to between 500 and 750°F, depending on coal calorific value and moisture content.

In theory, only the primary air must be heated; that is, air used to actually dry the coal in the pulverizers. Ignited fuel can burn without preheating the secondary and tertiary air. However, there is considerable advantage to the furnace heat-transfer process from heating *all* the combustion air; it increases the rate of burning and helps raise adiabatic temperature.

5.1.4 Fossil Fuels

Fossil fuels used for steam generation in utility and industrial power plants may be classified into solid, liquid, and gaseous fuels. Each fuel may be further classified as a natural, manufactured, or by-product fuel. Not mutually exclusive, these classifications necessarily overlap in some areas. Obvious examples of natural fuels are coal, crude oil, and natural gas.

Of all the fossil fuels used for steam generation in electric-utility and industrial power plants today, coal is the most important. It is widely available throughout much of the world, and the quantity and quality of coal reserves are better known than those of other fuels.

5.1.5 Classification of Coal

Coals are grouped according to rank. For the purposes of the power-plant operator, there are several suitable ranks of coal:

- Anthracite
- Bituminous
- Subbituminous
- Lignite

The following description of coals by rank gives some of their physical characteristics.

Anthracite. Hard and very brittle, anthracite is dense, shiny black, and homogeneous with no marks or layers. Unlike the lower-rank coals, it has a high percentage of fixed carbon and a low percentage of

TABLE 5-1 Classification of Coals of Rank*

Class and group	Fixed carbon limits, % (dry, mineral- matter-free basis)		Volatile matter limits, % (dry, mineral- matter-free basis)		Calorific value limits, Btu/lb (moist, [†] mineral-matter- free basis)		
	Equal to or greater than	Less than	Equal to or greater than	Less than	Equal to or greater than	Less than	Agglomerating character
Anthracitic							
Metaanthracite	98	...	...	2	...	...	Nonagglomerating
Anthracite	92	98	2	8	...	...	
Semianthracite‡	86	92	8	14	...	...	
Bituminous							
Low-volatile bituminous coal	78	86	14	22	...	...	Commonly glomerating¶
Medium-volatile bituminous coal	69	78	22	31	...	...	
High-volatile A bituminous coal	...	69	31	...	14,000§	...	
High-volatile B bituminous coal	...	...	...	...	13,000§	14,000	
High-volatile C bituminous coal	...	...	...	...	11,500 10,500	13,000 11,500	
Agglomerating							
Subbituminous							
Subbituminous A coal	...	...	...	...	10,500	11,500	Nonagglomerating
Subbituminous B coal	...	...	...	...	9,500	10,500	
Subbituminous C coal	...	...	...	...	8,300	9,500	
Lignitic							
Lignite A	...	...	...	...	6,300	8,300	
Lignite B	...	...	...	...	...	6,300	

Note: 1 Btu/lb = 2326 J/kg.

*This classification does not include a few coals, principally nonbanded varieties, which have unusual physical and chemical properties and which come within the limits of fixed carbon or calorific value of the high-volatile bituminous and subbituminous ranks. All of these coals either contain less than 48% dry, mineral-matter-free fixed carbon or have more than 15,500 moist, mineral-matter-free Btu per pound.

†Moist refers to coal containing its natural inherent moisture but not including visible water on the surface of the coal.

‡If agglomerating, classify in low-volatile group of the bituminous class.

§Coals having 69% or more fixed carbon on the dry, mineral-matter-free basis shall be classified by fixed carbon, regardless of calorific value.

¶It is recognized that there may be nonagglomerating varieties in these groups of the bituminous class, and there are notable exceptions in the high-volatile C bituminous group.

Source: ASTM Standards D388, *Classification of Coals by Rank*.

volatile matter. Anthracites include a variety of slow-burning fuels merging into graphite at one end and into bituminous coal at the other. They are the hardest coals on the market, consisting almost entirely of fixed carbon, with the little volatile matter present in them chiefly as methane, CH₄. Anthracite is usually graded into small sizes before being burned on stokers. The “metaanthracites” burn so slowly as to require mixing with other coals, while the “semianthracites,” which have more volatile matter, are burned with relative ease if properly fired. Most anthracites have a lower heating value than the highest-grade bituminous coals. Anthracite is used principally for heating homes and in gas production.

Some semianthracites are dense, but softer than anthracite, shiny gray, and somewhat granular in structure. The grains have a tendency to break off in handling the lump, and produce a coarse, sand-like slack. Other semianthracites are dark gray and distinctly granular. The grains break off easily in

handling and produce a coarse slack. The granular structure has been produced by small vertical cracks in horizontal layers of comparatively pure coal separated by very thin partings. The cracks are the result of heavy downward pressure, and probably shrinkage of the pure coal because of a drop in temperature.

Bituminous. By far the largest group, bituminous coals derive their name from the fact that on being heated they are often reduced to a cohesive, binding, sticky mass. Their carbon content is less than that of anthracites, but they have more volatile matter. The character of their volatile matter is more complex than that of anthracites, and they are higher in calorific value. They burn easily, especially in pulverized form, and their high volatile content makes them good for producing gas. Their binding nature enables them to be used in the manufacture of coke, while the nitrogen in them is utilized in processing ammonia.

The low-volatile bituminous coals are grayish-black and distinctly granular in structure. The grain breaks off very easily, and handling reduces the coal to slack. Any lumps that remain are held together by thin partings. Because the grains consist of comparatively pure coal, the slack is usually lower in ash content than are the lumps.

Medium-volatile bituminous coals are the transition from high-volatile to low-volatile coal and, as such, have the characteristics of both. Many have a granular structure, are soft, and crumble easily. Some are homogeneous with very faint indications of grains or layers. Others are of more distinct laminar structure, are hard, and stand handling well.

High-volatile A bituminous coals are mostly homogeneous with no indication of grains, but some show distinct layers. They are hard and stand handling with little breakage. The moisture, ash, and sulfur contents are low, and the heating value is high.

High-volatile B bituminous coals are of distinct laminar structure; the layers of black, shiny coal alternate with dull, charcoal-like layers. They are hard and stand handling well. Breakage occurs generally at right angles and parallel to the layers, so that the lumps generally have a cubical shape.

High-volatile C bituminous coals are of distinct laminar structure, are hard, and stand handling well. They generally have high moisture, ash, and sulfur contents and are considered to be free-burning coals.

Subbituminous. These coals are brownish black or black. Most are homogeneous with smooth surfaces, and with no indication of layers. They have high moisture content, as much as 15% to 30%, although appearing dry. When exposed to air they lose part of the moisture and crack with an audible noise. On long exposure to air, they disintegrate. They are free-burning, entirely noncoking, coals.

Lignite. Lignites are brown and of a laminar structure in which the remnants of woody fibers may be quite apparent. The word *lignite* comes from the Latin word *lignum* meaning wood. Their origin is mostly from plants rich in resin, so they are high in volatile matter. Freshly mined lignite is tough, although not hard, and it requires a heavy blow with a hammer to break the large lumps. But on exposure to air, it loses moisture rapidly and disintegrates. Even when it appears quite dry, the moisture content may be as high as 30%. Owing to the high moisture and low heating value, it is not economical to transport it long distances.

Unconsolidated lignite (B in Table 5-1) is also known as “brown coal.” Brown coals are generally found close to the surface, contain more than 45% moisture, and are readily won by strip mining.

5.1.6 Impact of Fuel on Boiler Design

The most important item to consider when designing a utility or large industrial steam generator is the fuel the unit will burn. The furnace size, the equipment to prepare and burn the fuel, the amount of heating surface and its placement, the type and size of heat-recovery equipment, and the flue-gas-treatment devices are all fuel dependent.

The major differences among those boilers that burn coal or oil or natural gas result from the ash in the products of combustion. Firing oil in a furnace results in relatively small amounts of ash; there is no ash from natural gas. For the same output, because of the ash, coal-burning boilers must have larger furnaces and the velocities of the combustion gases in the convection passes must be lower. In addition, coal-burning boilers need ash-handling and particulate-cleanup equipment that costs a great deal and requires considerable space.

TABLE 5-2 Representative Coal Analyses

	Medium-volume bituminous	High-volume bituminous	Subbituminous C	Low-sodium lignite	Medium-sodium lignite	High-sodium lignite
Total H ₂ O, %	5.0	15.4	30.0	31.0	30.0	39.6
Ash, %	10.3	15.0	5.8	10.4	28.4	6.3
VM, %	31.6	33.1	32.6	31.7	23.2	27.5
FC, %	53.1	36.5	36.6	26.9	18.4	26.6
Btu/lb, as fired	13,240	10,500	8,125	7,590	5,000	6,523
Btu/lb, MAF	15,640	15,100	12,650	12,940	12,020	12,050
Fusion (reducing), °F						
Initial def.	2,170	1,990	2,200	2,075	2,120	2,027
Softening	2,250	2,120	2,250	2,200	2,380	2,089
Fluid	2,440	2,290	2,290	2,310	2,700	2,203
Ash analysis, %						
SiO ₂	40.0	46.4	29.5	46.1	62.9	23.1
Al ₂ O ₃	24.0	16.2	16.0	15.2	17.5	11.3
Fe ₂ O ₃	16.8	20.0	4.1	3.7	2.8	8.5
CaO	5.8	7.1	26.5	16.6	4.8	23.8
MgO	2.0	0.8	4.2	3.2	0.7	5.9
Na ₂ O	0.8	0.7	1.4	0.4	3.1	7.4
K ₂ O	2.4	1.5	0.5	0.6	2.0	0.7
TiO ₂	1.3	1.0	1.3	1.2	0.8	0.5
P ₂ O ₅	0.1	0.1	1.1	0.1	0.1	0.2
SO ₃	5.3	6.0	14.8	12.7	4.6	17.7
Sulfur, %	1.8	3.2	0.3	0.6	1.7	0.8
Lb H ₂ O/million Btu	3.8	14.7	36.9	40.8	60.0	60.7
Lb ash/million Btu	7.8	14.3	7.1	13.7	56.8	9.7
Fuel-fired,*1000 lb/h	405	520	705	750	1,175	900

Note: 1 Btu/lb = 2326 J/kg; $t_{\text{FC}} = (t_{\text{FF}} - 32)/1.8$; 1 lb = 0.4536 kg; 1 Btu = 1055 J.

*Constant heat output, nominal 600-MW unit, adjusted for efficiency.

Table 5-2 lists the variation in calorific values and moisture contents of several coals, and the mass of fuel that must be handled and fired to generate the same electrical-power output. These values are important because the quantity of fuel required helps determine the size of the coal-storage yard, as well as the handling, crushing, and pulverizing equipment for the various coals.

Furnace Sizing. The most important step in coal-fired unit design is to properly size the furnace. Furnace size has a first-order influence on the size of the structural-steel framing, the boiler building and its foundations, as well as on the sootblowers, platforms, stairways, steam piping, and duct work. The fuel-ash properties that are particularly important when designing and establishing the size of coal-fired furnaces include

The ash fusibility temperatures (both in terms of their absolute values and the spread or difference between initial deformation temperature and fluid temperature)

The ratio of basic to acidic ash constituents

The iron/calcium ratio

The fuel-ash content in terms of pounds of ash per million British thermal units

The ash friability

These characteristics and others translate into the furnace sizes in Fig. 5-5, which are based on the six coal ranks shown in Table 5-2. This size comparison illustrates the philosophy of increasing the furnace plan area, volume, and the fuel burnout zone (the distance from the top fuel nozzle to the furnace arch), as lower-grade coals with poorer ash characteristics are fired.

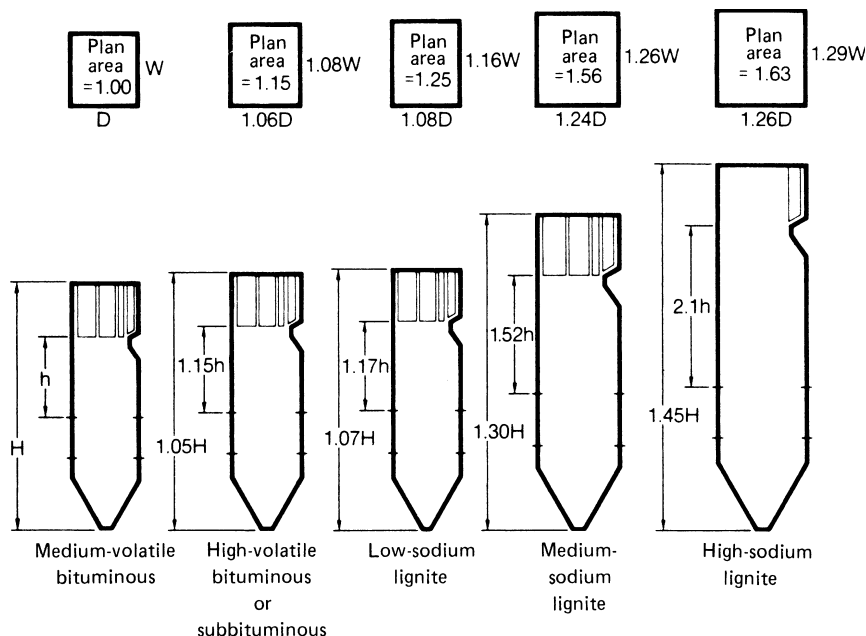


FIGURE 5-5 Effect of coal rank on furnace sizing (constant heat output).

Figure 5-5 is a simplified characterization of actual furnaces built to burn the fuels listed in Table 5-2. Wide variations exist in fuel properties within coal ranks, as well as within several subclassifications (e.g., subbituminous A, B, C), each of which may require a different size furnace.

Among the most important design criteria in large pulverized-fuel furnaces are net heat input in British thermal units per hour per square foot of furnace plan area (NHI/PA) and the vertical distance from the top fuel nozzle to the furnace arch. Furnace dimensions must be adequate to establish the necessary furnace retention time to properly burn the fuel as well as to cool the gaseous combustion products. This is to ensure that the gas temperature at the entrance to the closely spaced convection surface is well below the ash-softening temperature of the lowest-quality coal burned. Heat-absorption characteristics of the walls are maintained using properly placed wall blowers to control the furnace outlet gas temperature by removing ash deposited on the furnace walls below the furnace outlet plane.

5.1.7 Environmental Considerations

Concerns for the control of air quality have probably had the largest single impact on power plant site selection, design, operation, and cost. The three classes of emissions which are of major concern are nitrogen oxides, sulfur oxides, and particulate matter.

Nitrogen Oxides. In the United States, nitrogen oxides can be controlled within federal, state, and local regulatory limits by in-furnace and postcombustion techniques. With respect to firing systems, each steam-generator manufacturer has developed specific design concepts for reducing nitrogen oxides. The common characteristics of all of these designs, however, included a careful regulation of the fuel/air ratio in the firing zone where the major fraction of the fuel nitrogen compounds are liberated and control of the heat-liberation pattern in the furnace. Postcombustion reduction methods utilizing reagents with or without catalysts are somewhat similar in concept among the steam-generator suppliers.

Particulate Control. The traditional particulate control device in power plant applications has been the electrostatic precipitator. In recent years, fabric filters (also called “baghouses”) have become increasingly popular.

In electrostatic precipitation, suspended particles in the gas are electrically charged, then driven to collecting electrodes by an electrical field; the electrodes are rapped to cause the particles to drop into collecting hoppers. This process differs from mechanical or filtering processes in which forces are exerted directly on the particulates rather than the gas as a whole. Effective separation of particles can be achieved with lower power expenditure, with negligible draft loss, and with little or no effect on the composition of the gas.

The principle of electrostatic precipitation is relatively simple. The process applies an electrostatic charge to dust particles with a corona discharge and passes them through an electric field where the particles are attracted to a collecting surface. The basic elements of a precipitator include a source of unidirectional voltage, corona or discharge electrodes, collecting electrodes, and a means of removing the collected matter.

Single-stage (Cottrell-type) precipitators combine the ionizing and collecting step. In the more common plate type, the electrodes are suspended between plates on insulators connected to a high-voltage source. A voltage differential created between the discharge and collecting electrodes develops a strong electric field between them. The flue gas is passed through the field and a unipolar discharge of gas ions, from the discharge electrode, is attached to the particulate matter.

5.1.8 Fabric Filtration

Fabric filters, or baghouses, have a long history of applications in both dry and wet filtration processes to recover chemicals or control stack emissions. Available materials limited early baghouse installations to temperatures below 250°F, and air dilution was frequently used ahead of the baghouse. In addition, the chemical-resistance characteristics of the bags also curtailed fabric filtration. These two limitations retarded its development for many years, particularly as available precipitator equipment met the existing regulations.

Serious consideration of this technology began after 1970; interest heightened as installations on large coal-fired boilers demonstrated good operating characteristics and high particulate-removal efficiencies.

5.1.9 Flue-Gas Desulfurization Systems

Flue-gas desulfurization (FGD) began in England in 1935. The technology remained dormant until the mid-1960s when it became active primarily in the United States and Japan. Since then, over 50 FGD processes have been developed, differing in the chemical reagents and the resultant end products.

The most common FGD system is a lime/limestone wet scrubber. After the flue gas has been treated in the precipitation (or baghouse), it passes through the induced fans and enters the SO₂ scrubber. If the required SO₂ removal efficiency is less than 85%, a fraction of the flue gas can be treated while bypassing the rest to mix with and reheat the saturated flue gas leaving the scrubber.

For higher-sulfur fuels requiring SO₂ removal efficiencies of 90% or greater, the entire flue-gas stream must be treated. Upon leaving the SO₂ absorption section, the flue gas is passed through entrainment separators to remove any slurry droplets mixed with the gas. The saturated flue gas is then reheated approximately 25 to 50°F above the water dewpoint before it is vented to the stack.

For low- to medium-sulfur fuels, an alternate scrubbing technology is dry scrubbing. This process minimizes water consumption and eliminates the requirement for flue-gas reheating but requires more expensive additives than the wet limestone systems.

The typical dry SO₂ absorber is a cocurrent classifying spray dryer. Flue gas enters the top of the absorber through inlet assemblies containing swirl vanes. The absorbent is injected pneumatically into the center of each swirler assembly by ultrasonic atomizing nozzles that require an air pressure of about 60 lb/in² (gauge). Slurry feed pressures are 10 to 15 lb/in² (gauge). The compressed air induces

primary dispersion of the absorbent slurry by mechanical shear forces produced by the two fluid streams. Final dispersion is accomplished by shattering the droplets with ultrasonic energy produced by the compressed air used with a proprietary nozzle design. Then ultrasonic nozzles generate extremely fine droplets, which have diameters that range from 10 to 50 μm , as shown by photographic studies.

The flue-gas outlet design requires that effluent gases make a 180° turn before leaving the absorber. Besides eliminating product accumulation in the outlet duct, the abrupt directional change also allows the larger particles to drop out in the absorber product hopper. This design curtails the particulate loading to the fabric filter. Consequently, the number of cleaning cycles as well as abrasion of the filter medium are reduced.

As compared with ordinary fly-ash collection applications, fabric filters together with dry scrubbing offer a broader choice of design options. In conventional fly-ash collection applications, the fabric filter experiences flue-gas temperatures about 100 to 150°F higher than encountered in dry scrubbing. Filter media unsuitable at the higher temperatures can be used when the fabric filter follows a dry absorber. In particular, acrylic fibers become attractive because of their strength and flex characteristics, as well as their ability to support more vigorous cleaning methods like mechanical shaking.

5.1.10 Advanced Methods of Using Coal

Coal, which is the most abundant and economically stable fossil fuel in the United States, continues to grow in use while under pressure to meet the most stringent federal and local emissions requirements. This trend has added to the cost and complexity of coal combustion technologies.

Emission-control methods that facilitate the use of coal in power plants can be classified as

- Precombustion processes

- In situ combustion processes

- Postcombustion processes

Precombustion processes include methods to clean the coal of sulfur-bearing compounds by wet separation, coal gasification, and coal liquefaction techniques. Coal gasification involves the partial oxidation of coal to produce a clean gas or by production of a “clean fuel” through coal liquefaction. Sulfur and ash are removed in these processes. The use of coal to produce a gas is not a new idea; it has been used to produce “town gas” for over 200 years. But its use in the United States had almost disappeared by 1930, because natural gas was abundant and low in cost. Concerns about the availability and economic stability of gas supplies, along with environmental trends, have renewed interest in coal gasification to produce substitute natural gas (SNG) and low- and medium-heat-content (LBTU and MBTU) gas for chemical feedstock or power plant fuel. Coal gasification in the combined-cycle mode has been well established as a viable technology for producing power with very low emissions both in the United States and Europe. New plants are using technologies such as high-temperature gas turbines, hot-gas cleanup to remove 99% of the sulfur (H_2S), and higher-pressure combined steam cycles to achieve overall efficiencies of greater than 40%. New integrated gasification combined-cycle (IGCC) plants of as much as 250 MWe are available. IGCC technology produces very low emissions per kilowatt of power and is therefore very attractive for the production of power. Likewise, coal liquefaction is not a new technology, but is only in limited commercial use in the United States. South Africa is the largest producer of synthetic liquid fuels from coal. Large-scale production of synthetic liquid fuels from coal began in 1910 in Germany with the Fischer-Tropsch process, which is used to produce a variety of fuels.

In fluidized-bed combustion, an in situ combustion-emission-control process, 90% to 95% of the SO_2 is captured during combustion by a sorbent (limestone). In this process, the NO_x production is low because of the low temperature at which the combustion reaction takes place. NO_x levels well fired below 0.25 lb/MBtu have been achieved with certain coals. Fluidized-bed combustion was developed in the 1950s and is now available for electric power plants of up to 300-MWe size. The technology has

three distinct types of units: bubbling bed, hybrid velocity, and circulating fluidized bed (CFB). CFB technology is the most popular fluidized-bed process and has evolved as a low-emission technology with excellent fuel flexibility for the production of power. Bubbling and hybrid-velocity fluidized-bed technologies have demonstrated low emissions while burning low-rank coals, waste fuels such as petroleum coke, and renewable fuel such as wood and peat. Hybrid-velocity fluidized-bed combustion can be readily retrofit to many older boilers that need pollution-control technology. Pressurized fluidized-bed combustion is used to achieve low sulfur and NO_x emissions of fluidized-bed combustion integrated with a gas turbine to achieve high cycle efficiency, and therefore make more efficient use of coal.

Postcombustion control processes are widely used for the capture of sulfur and particulate. Lime-based scrubbers for SO_2 removal and equipment for particulate control were described in Sec. 5.1.9.

Processes and equipment for removal of NO_x from flue gases leaving boilers have been widely used in Europe and are being applied in the United States. In situ control of NO_x by modifications to firing technology and over-fire air can reduce NO_x as much as 50%. Selective noncatalytic control (SNCR) involves ammonia or urea sprayed in the proper place in the boiler to reduce NO_x . More NO_x reduction can be achieved by selective catalytic reduction (SCR), which uses ammonia in a postcombustion control system. SCR can reduce NO_x levels well below those from a conventional pulverized-coal boiler.

Coal gasification is an efficient way to produce electric power while minimizing the emissions from the combustion of coal. Coal gasification can achieve cycle efficiencies above 40% when the gas turbine cycle is completely integrated with the steam cycle. This is referred to as the *integrated gasification combined cycle* (IGCC) (Fig. 5-6). In an IGCC plant, the gas from the gasification process is burned in a boiler or gas turbine for the generation of electric power. The process also uses the heat from the gas turbine exhaust to produce electric power from a steam cycle.

In the gasification process, coal is partially reacted with a deficiency of air to produce low-heating-value fuel gas. The gas is cleaned of particulate and then sulfur compounds in a hot-gas cleanup system. Elemental sulfur is disposed of or sold.

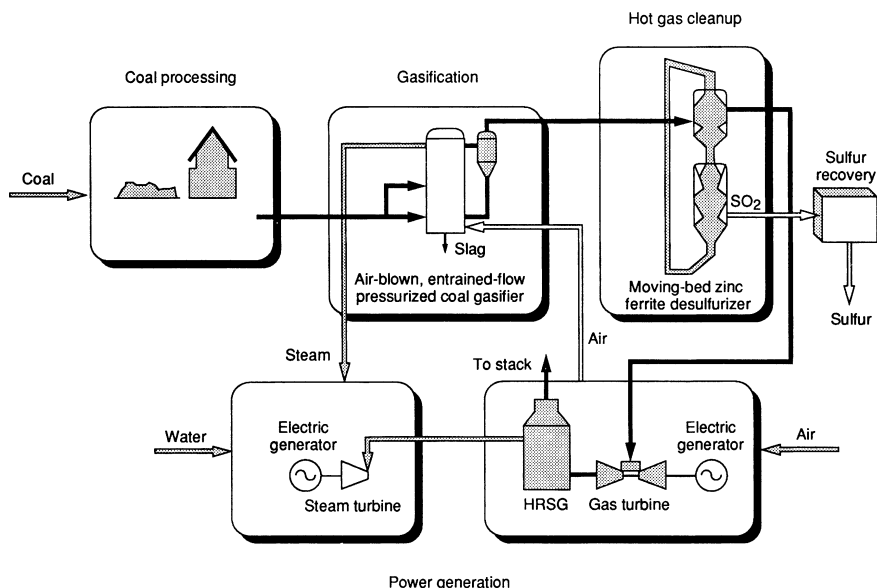


FIGURE 5-6 Integrated combined-cycle power plant.

5.1.11 Fluidized-Bed Combustion

For decades, fluidized-bed reactors have been used in noncombustion reactions in which the thorough mixing and intimate contact of the reactants in a fluidized bed result in high product yield with improved economy of time and energy. Although conventional methods of burning coal can also generate energy with very high efficiency, fluidized-bed combustion can burn coal efficiently at a temperature low enough to avoid many of the problems of conventional combustion.

The outstanding advantage of fluidized-bed combustion (FBC) is its ability to burn high-sulfur coal in an environmentally acceptable manner without the use of flue-gas scrubbers. A secondary benefit is the formation of lower levels of nitrogen oxides compared to other combustion methods.

5.1.12 Circulating Fluidized-Bed Steam Generators

Figure 5-7 shows a typical CFB steam generator. Crushed fuel and sorbent are fed mechanically or pneumatically to the lower portion of the combustor. Primary air is supplied to the bottom of the combustor through an air distributor, with secondary air fed through one or more elevations of air ports in the lower combustor. Combustion takes place throughout the combustor, which is filled with bed material. Flue gas and entrained solids leave the combustor and enter one or more cyclones where the solids are separated and fall to a seal pot. From the seal pot, the solids are recycled to the combustor. Optionally, some solids may be diverted through a plug valve to an external fluidized-bed heat exchanger (FBHE) and back to the combustor. In the FBHE, tube bundles absorb heat from the fluidized solids.

Bed temperature in the combustor is essentially uniform and is maintained at an optimum level for sulfur capture and combustion efficiency by heat absorption in the walls of the combustor and in the FBHE (if used). Flue gas leaving the cyclones passes to a convection pass, air heater, baghouse, and induced-draft (ID) fan. Solids inventory in the combustor is controlled by draining hot solids through an ash cooler.

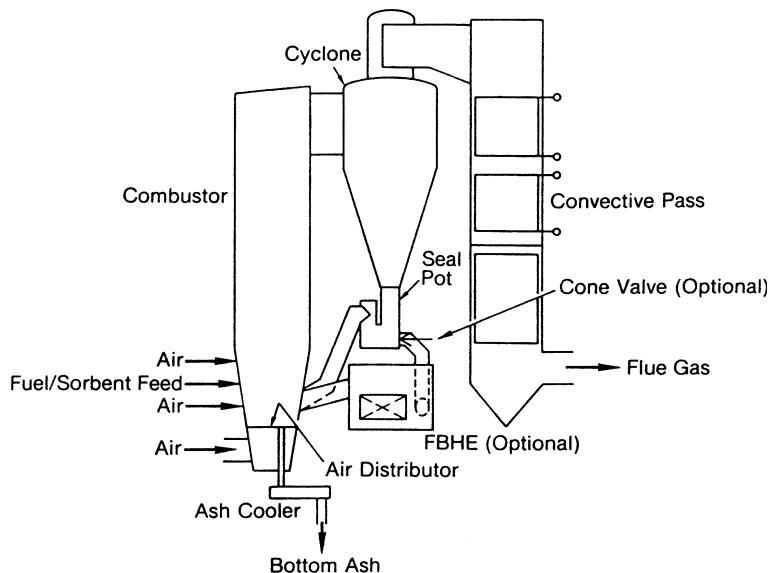


FIGURE 5-7 Typical circulating fluidized-bed (CFB) steam generator.

5.2 NUCLEAR POWER PLANTS

By GEORGE H. MILEY

5.2.1 Nuclear Energy

Introduction. The United States is the world's largest supplier of commercial nuclear power. In 2005, there were 104 U.S. commercial nuclear generating units that were fully licensed to operate. One reactor, however, Brown's Ferry unit 1 has been shut down since 1985. Therefore, some sources cite only 103 units. Together, they provide about 20% of the nation's electricity—second only to coal as a fuel source.

The Energy Information Administration (EIA) reports that the U.S. nuclear industry generated 788,556 million kilowatt hours of electricity in 2004 (Fig. 5-8), a new U.S. (and international) record. Although no new U.S. nuclear power plants have come on line since 1996, this is the industry's fifth annual record since 1998.

General. Applying the nuclear process for electrical production involves consideration of characteristics substantially different from those associated with the use of fossil fuels. With fossil fuels or with hydro, the amount of energy source (fuel) supplied to the power plant is proportional to the power demanded at that time. With nuclear power, however, the fuel for a substantial amount of energy output is physically located in the converter at any time. A second important characteristic of the nuclear process is the energy density. The thermal energy density in a typical fossil boiler (heated volume or core volume) is in the range of 0.20 kW/L; in a typical nuclear power generator it is in

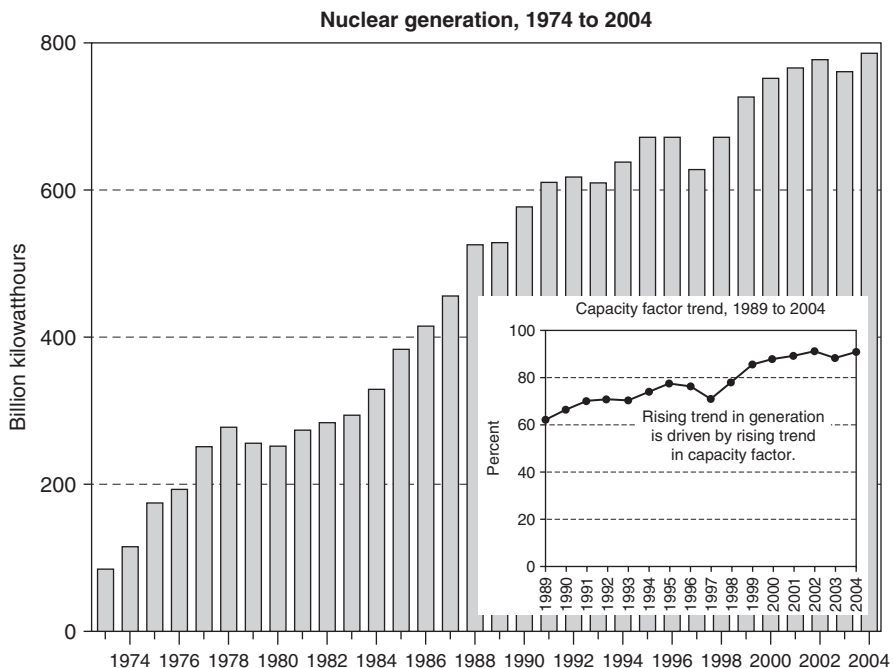


FIGURE 5-8 U.S. nuclear power generation. (Source: Energy Information Administration, *Monthly Energy Review*.)

the range of 80 kW/L. A third important difference is that of continued low-level heat generation (decay heat) when the nuclear process is shut down following power operation. A fourth important difference is that of emanations. The fossil process requires the intake of large volumes of air and fuel and the corresponding exhaust of large volumes of waste gas, including CO₂, SO₃, NO₂, etc., some particulate matter, and in the case of coal-fired boilers, substantial quantities of ash. The nuclear process, however, requires only the input of the material placed in the core; its output is fuel-element materials plus radioactive “waste” products from the fission process. This residue includes small quantities of gases which may be released or may be stored and solids which are contained within the fuel.

These and other more subtle aspects introduce many new considerations in the equipping and regulation of the nuclear process.

5.2.2 Mass-Energy Relationships

One of the first applications of the special theory of relativity proposed by Einstein in 1905 was the interrelation between mass and energy, expressed by the equation $E = mc^2$. Thus, a change in nuclear mass appears as energy. If the mass m is expressed in kilograms and the velocity of light c in meters per second, the energy E is in joules.

$$\begin{aligned} E(\text{J}) &= \text{mass}(\text{kg}) \times (2.998 \times 10^8 \text{ m/s})^2 \\ &= \text{mass}(\text{kg}) \times 8.99 \times 10^{16} \text{ m}^2/\text{s}^2 \end{aligned} \quad (5-1)$$

The amounts of energy involved in single nuclear events are usually very small. Thus, for convenience, the electronvolt (the energy acquired by any charged particle carrying a unit electronic charge falling through a potential of 1 V) is often used. One electronvolt (eV) = 1.602×10^{-19} J and, correspondingly, 1 keV = 1.602×10^{-16} J. One MeV = 1.602×10^{-13} J.

The mass-energy relationships become

$$\begin{aligned} E(\text{eV}) &= \text{mass}(\text{kg}) \times \frac{8.99 \times 10^{16} \text{ m}^2/\text{s}^2}{1.602 \times 10^{-19} \text{ J/eV}} \\ &= \text{mass}(\text{kg}) \times 5.61 \times 10^{35} \text{ eV/kg} \\ E(\text{keV}) &= \text{mass}(\text{kg}) \times 5.61 \times 10^{32} \text{ keV/kg} \\ E(\text{MeV}) &= \text{mass}(\text{kg}) \times 5.61 \times 10^{29} \text{ MeV/kg} \end{aligned}$$

where $1 \text{ J} = 1 \text{ m}^2 \cdot \text{kg}/\text{s}^2$.

It is often convenient to use the energy corresponding to 1 atomic mass unit (amu). One amu = 1.657×10^{-27} kg (1 amu = $1/12$ of the mass of a neutral atom of ¹²C).

$$\begin{aligned} E_{\text{amu}} &= 1.66 \times 10^{-27} \text{ kg} \times 5.61 \times 10^{29} \text{ MeV/kg} \\ &= 931 \text{ MeV/amu} \end{aligned} \quad (5-2)$$

The atomic mass of a nuclide can be evaluated in terms of the masses of its constituent particles and the *binding energy* (Fig. 5-9). The mass of the nuclide is less than the sum of its constituent particles in the free state. If ΔM is the decrease in mass when a number of protons, neutrons, and electrons combine to form an atom, then the mass-energy equivalence principle states that an amount of energy equal to $\Delta E = c^2 \Delta M$ is released in the process. The difference in mass ΔM is called the *mass defect*; it is the amount of mass which would be converted to energy if a particular atom or nuclide were to be assembled from the requisite number of protons, neutrons, and electrons. The same amount of energy would be needed to break the atom into its constituent particles, and the energy equivalent of the mass defect is therefore a measure of the binding energy of the nuclei. The mass of

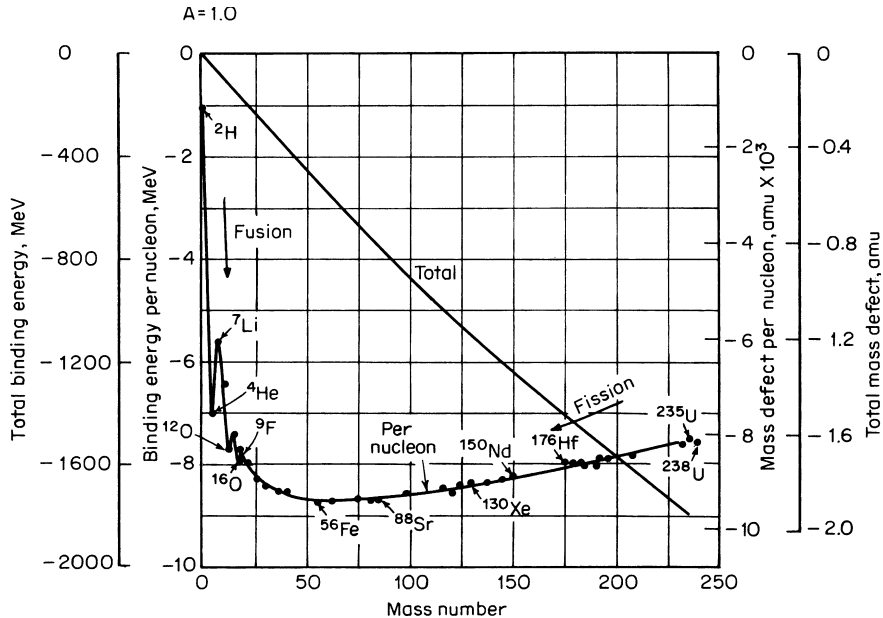


FIGURE 5-9 Mass defects and binding energies of nuclei.

the constituent particles is the sum of Z proton masses, Z electrons, and $A - Z$ neutrons, where A refers to the mass number of the element. Pairs of protons and electrons can be represented by hydrogen atoms; the loss in mass which accompanies the formation of the hydrogen atom from the proton and an electron is negligible. The mass defect can then be written $\Delta M = ZM_H + (A - Z)M_n - M_{ZA}$, where M_H is the mass of the hydrogen atom, 1.008142 amu; M_n is the mass of the neutron, 1.008982 amu; and M_{ZA} is the mass of nuclide of concern.

Figure 5-9 provides an approximate picture of the nuclear binding energy. In the higher mass numbers, the actual binding energy is not the same for each particle in the nucleus. After the maximum of the curve, almost every successive particle (proton or neutron) is bound less tightly than those already present, and the overall average decreases. The binding energy represented, however, is sufficiently accurate for engineering evaluations.

5.2.3 The Fission Process

In the higher mass numbers, several of the naturally occurring elements are radioactive or have a characteristic which enables them to emit nuclear particles and be transmuted to different elements as a function of time. The various naturally occurring series are designated the thorium, uranium, and actinium series. These designations are related to the elements at or near the head of the series and can be expressed as multiples of a number N , where N is an integer. The series are indicated by $4N$, $4N + 2$, and $4N + 3$, respectively. There is no naturally occurring $4N + 1$ element; such an element has been created in the process of artificial nuclear transmutation. This element is designated neptunium and has the mass characteristic of $4N + 1$. It, too, heads a radioactive series. The four radioactive series are shown in Fig. 5-10.

A number of elements with high mass numbers, both natural and artificially produced, undergo a process of nuclear fission. In the fission process, a nucleus absorbs a neutron and the resulting compound nucleus is so unstable that it immediately breaks up into parts. As shown by the arrow labeled "fission" in Fig. 5-9, the fission products have a lower mass and larger binding energy, resulting in

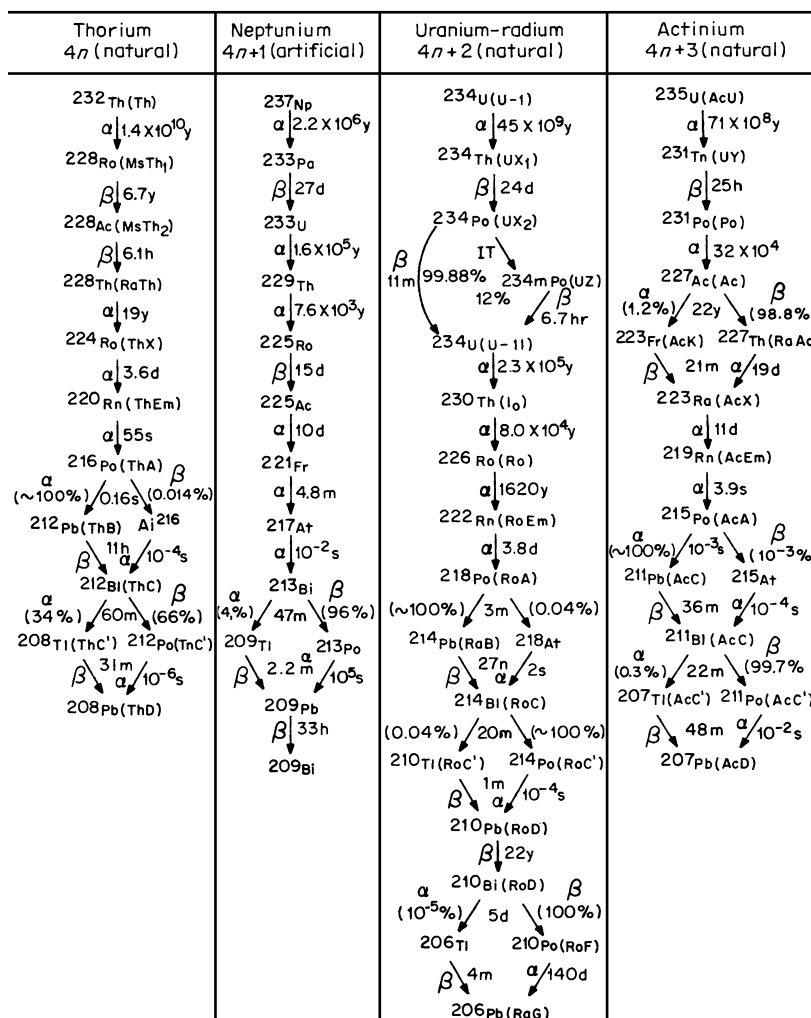


FIGURE 5-10 The four radioactive series.

a release of energy in the form of kinetic energy of the products. (Also note in Fig. 5-9 that the arrow for “fusion” shows lighter elements fusing together to create higher mass products, again with a release of energy by emission of high-speed products. Many of the heavy nuclides can be induced to fission, but most only with neutrons of high energy. Naturally occurring heavy nuclides that fission with neutrons of energy in the range of the neutrons produced by the fission are uranium isotopes ^{235}U and ^{238}U and thorium 232. In addition, artificially produced nuclides ^{233}U and ^{239}Pu , produced by (n, β) reactions in ^{232}Th and ^{238}U , respectively, are capable of fission. The fission process, in a nuclear reactor, is initiated by neutrons which are generated as part of the process. The general fission process may be expressed by

$$^m\text{F} + {}^1_0\text{n} \rightarrow {}^x\text{A} + {}^{m-(x+C)}\text{B} + {}^C{}_1\text{n} \quad (5-3)$$

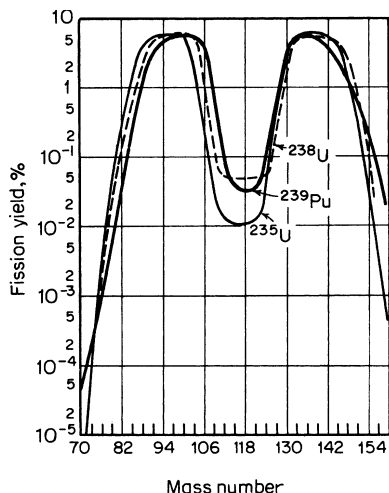


FIGURE 5-11 Fission yield.

TABLE 5-3 Distribution of Fission Energy

Energy	MeV
Kinetic energy of fission fragments	168 ± 5
Instantaneous gamma-ray energy	5 ± 1
Kinetic energy of fission neutrons	5 ± 0.5
Beta particles from fission products	7 ± 1
Gamma rays from fission products	6 ± 1
Neutrinos	~ 10
Total fission energy	201 ± 6

ments. Table 5-3 shows the distribution of fission energy. For practical purposes, the neutrino energy, because of the low probability of interaction of neutrinos with matter, is not recoverable. (This leaves about 190 MeV, or 3.0×10^{-11} J, recoverable per fission.)

5.2.4 Neutron Interaction

Each neutron interacting with a nucleus does not always result in fission; some are scattered and some are involved in radiative capture, that is, initiate the radiation of other particles and/or photons to reduce the target atom to a stable state. The neutron-absorption probability for a given nuclide is referred to as its cross section and is expressed in units of area. Since very small areas are involved, the special unit for cross section is the barn, equal to 10^{-24} cm².

The cross section for a neutron interaction varies with energy. An explanation is that, quantum mechanically, the wavelength λ of the neutron is inversely proportional to its energy E or velocity, and may be expressed by

$$\lambda = \frac{2.86 \times 10^{-8}}{\sqrt{E(\text{eV})}} \quad \text{mm} \quad (5-5)$$

where F = fuel nuclide, mass number m

n = neutron

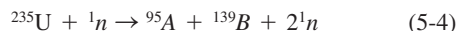
A, B = fragment nuclides

C = number of neutrons produced

x = atomic number

The percentage of nuclide production as a function of mass number is shown in Fig. 5-11.

A typical example is the fission of ^{235}U with the production of two most likely fission fragments.



The mass balances of this equation are

Before fission	After fission
$^{235}\text{U} - 235.124$ amu	$^{95}\text{A} - 94.945$ amu
${}^1_0\text{n} - 1.009$ amu	$^{139}\text{B} - 138.955$ amu
	$2{}^1_0\text{n} - 2.018$ amu
236.133 amu	235.918 amu

The mass change resulting from fission is $236.133 - 235.918 = 0.215$ amu, which by the relationship of mass to energy is equivalent to $E(J) = \text{mass}(\text{amu}) \times 1.49 \times 10^{-10}$ J/amu, which represents $\sim 3.2 \times 10^{-11}$ J/fission or approximately 200 MeV/fission (or 3.2×10^{-11} W · s/fission).

The major portion of this energy is released immediately as kinetic energy of the fission fragments, the fission neutrons, and instantaneous gamma rays. A portion of the energy is released gradually from the decay of the fission frag-

For fast neutrons (about 1 MeV), λ is of the order of 10^{211} mm, and for thermal neutrons (about 0.03 MeV), λ is about 1.7×10^{27} mm. The slower neutrons behave as though they had a diameter approaching that of the atom, and thus have a larger probability of interaction.

5.2.5 Radiation

Nuclide Composition. The elements of the periodic table, both naturally occurring and artificial, are composed of protons (except for hydrogen), neutrons, and electrons. Many of the elements have two or more isotopic forms, states which have the same atomic number but a different atomic mass because of a different number of neutrons in the nucleus.

Most of the naturally occurring elements are stable, that is, do not eject particles to change to a different isotope or a different element. However, some naturally occurring elements, as indicated in Fig. 5-10, are conditionally stable and have a probability for transmutation. Out of the total number of atoms present, the probability indicates that a certain number of the atoms will, by ejecting a particle, change to an isotope or a new element. The mode of decay for a given isotope is predictable. The pattern is sometimes complex and follows a decay chain.

Radioactive Transmutation. For every radioactive material, there are characteristic quantities that may be used to describe the process. Each radioactive nuclide has a definite probability of decaying in unit time. This decay probability has a constant value, characteristic of the particular radioisotope. In a given sample, the rate of decay at any instant is proportional to the number of radioactive atoms present at that time. If N is the number of radioactive atoms present at time t and λ is the decay constant, the decay rate is given by $dn/dt = -\lambda N$ for a simple decay scheme. Integrating this over the interval N_0 to N gives

$$N = N_0 e^{-\lambda t} \quad (5-6)$$

where N = number of atoms remaining unchanged at any time t

N_0 = initial number of atoms

λ = disintegration constant

The reciprocal of the decay constant $1/\lambda$ is the mean or average life of the radioactive species = t_m . A more widely used quantity for quantifying radioactive decay is the half-life, that period of time during which half the atoms originally present are transmuted. If N is set equal to $1/2 N_0$ and the above equation is solved for t , the value becomes

$$t_{1/2} = \frac{\ln 2}{\lambda} = \frac{0.6931}{\lambda} \quad (5-7)$$

In a radioactive species, a nuclide may undergo successive decay before reaching the ground state. For a compound decay scheme involving two states A and B , the net rate of change of B with time is given by

$$\frac{dN_b}{dt} = \lambda_A N_A - \lambda_B N_B \quad (5-8)$$

where the solution is $N = [\lambda_A N_{A0}/(\lambda_B - \lambda_A)] (e^{-\lambda_A t} - e^{-\lambda_B t})$. The first term on the right represents the production of B from the decay of A ; the second term is the decay of B . N_{A0} is the number of parent atoms at time $t = 0$. Sample decay curves in Fig. 5-12 show both a simple decay and a compound two-stage decay.

If the radiation occurs by the emission of a quantity of energy (photon), the nuclide retains its atomic weight and number. If the decay occurs by emission of a particle, the nuclide changes to an isotope (same atomic number), an isobar (same mass number), or a different element.

Artificial elements, including those resulting from the fission process, are very likely to be radioactive. In some cases, this activity results in the emission of a photon of energy to allow the atom to reach a lower energy state. In other cases, a particle is emitted; the particle emitted for some decaying nuclides is a neutron. These delayed neutrons are important to the regulation of the fission process.

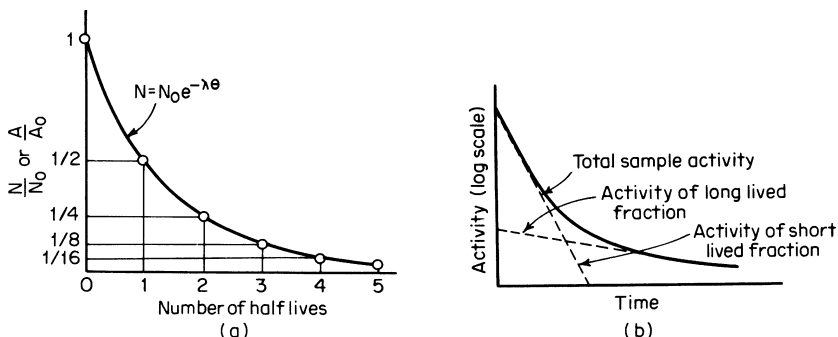


FIGURE 5-12 (a) Radioactive decay of a single radionuclide as a function of half-life; (b) decay of mixture of independent radionuclides.

Types of Radiation. There are three categories of radiation emanations of biological concern in nuclear power. The first category is that of charged particles, principally alpha particles and beta particles. The second is that of uncharged particles, chiefly neutrons. The third is that of photons or gamma rays. The charged particles directly produce ionization by collision with neutral atoms. Neutrons and photons indirectly produce ionization by liberating directly ionizing particles or by initiating nuclear transformations.

In radioactivity, a conventional unit is the curie, that quantity of any radioactive material giving 3.7×10^{10} disintegrations/s. For small quantities of radiation, the millicurie and the microcurie, 3.7×10^7 and 3.7×10^4 disintegrations/s, respectively, are frequently used. The rutherford (rd), equal to 10^6 disintegrations/s, is sometimes used. The SI unit of radioactivity is the becquerel [1 curie (Ci) = 3.7×10^{10} becquerel (Bq)].

The radioactivity, the decay constant, and the weight are related by

$$\frac{dN}{dt} = \lambda N = \frac{\lambda W_A}{G_w} \quad (5-9)$$

where λ = decay constant, disintegrations/s

W = weight of the material, g

A = Avogadro's number = 6.02×10^{23} atoms/mol

G_w = gram atomic weight of the material, g/mol

N = number of atoms

This equation shows that a given amount of radioactivity may occur from a large mass with a small decay rate or a small mass which has a high decay rate.

Radiation dosage is expressed in four ways:

1. **Absorbed dose (D),** which is the energy absorbed per unit mass at a specific place in a material. The standard of absorbed dose is the gray; $1 \text{ Gy} = 1 \text{ J/kg}$. The special unit of absorbed dose is the rad = $0.01 \text{ J/kg} = 0.01 \text{ Gy}$. A subset is the absorbed-dose index, which is the maximum absorbed dose, at a point, within a 300-mm-diameter sphere centered at the point and consisting of material equivalent to soft tissue with a density of 1 g/cm^3 .
2. **Dose equivalent (H).** In general, the biological equivalent of a given absorbed dose depends on the type of radiation and the irradiation conditions. The product of modifying factors, assigned to weigh the effect on a given organ, and the absorbed dose is the dose equivalent. The special unit of H is the rem (where D is in rads, H is in rems). A subset of this is the dose-equivalent index, which is the maximum dose equivalent, at a point, within a 300-mm-diameter sphere centered at the point and consisting of material equivalent to soft tissue with a density of 1 g/cm^3 .

3. *Kerma* (K), which is the sum of the initial kinetic energies of the charged particles produced by indirectly ionizing radiation per unit mass of the material in which the interaction takes place. The units of K are grays or rads.
4. *Exposure* (X) is the measure of a particular field of electromagnetic radiation (x- or gamma rays) to ionize air. The special unit of exposure is the roentgen (R) = 2.58×10^{-4} coulombs (C)/kg of air.

5.2.6 Nuclear Plant Safety

The nuclear-powered steam supply system characteristics of substantial energy potential present in the reactor, radiation production during the fission process, and continued radiation production and heat generation after shutdown require that special safety precautions be taken in design and operation of a nuclear plant. The health and welfare of the public depends on both the continuation of the plant's power production and the avoidance of any incident which would endanger the environment. In order to achieve the latter goal and to aid the former, special regulations relating to nuclear plants have been formulated.

Workers in power plants are covered by federal regulations, with special attention being devoted to radiation protection. A guiding principle applied throughout the industry is known as ALARA. This directs management and workers to seek *as low an exposure as is reasonably achievable*. Application of ALARA requires careful planning and a balance between minimizing exposure versus work requirements.

5.2.7 Federal Regulations

Title 10 of the Code of Federal Regulations (10 CFR) has the following parts which are of primary importance to nuclear power facilities:

- Part 20, Standards for Protection against Radiation
- Part 50, Licensing of Production and Utilization Facilities
- Part 55, Operators' Licenses
- Part 70, Special Nuclear Material
- Part 100, Reactor Site Criteria

There are several other parts of 10 CFR which relate to the usage or handling of radioactive material. Most of the parts previously listed have appendixes which treat requirements for specific subjects. Authority for regulation of commercial, nuclear-powered plants is vested with the U.S. Nuclear Regulatory Commission. This authority includes the licensing of new facilities and the surveillance of operating facilities. An applicant for a nuclear-powered plant is required to apply for a license to construct and operate the facility. Such application includes the submission of Safety Analysis Reports which describe the design bases, the design, and the analyses performed to show that plant performance and conditions will be within established limits.

5.2.8 Standards

Appendix A of 10 CFR Part 50 provides general design criteria for nuclear power plants. Criterion 1 requires that structures, systems, and components important to safety be designed, fabricated, erected, and tested to quality standards. The nuclear standards program of the American National Standards Institute has developed a sizable group of standards for this requirement. The principal design, systems, and operation standards are those developed by ASME, IEEE, ANS, and ISA.

Many other documents providing criteria, standard practices, or guidance are available. In the nuclear area, specific designs have not been repeated frequently enough to accumulate a significant backlog of experience. As a result, many of the "standards" are developed to provide leadership in addressing given areas.

The Nuclear Regulatory Commission provides guidance in many areas of design, construction, and operation through Regulatory Guides. Individual guides may cite a standard as an acceptable method of addressing the area concerned.

5.2.9 Quality Assurance

The best defense against incidents which endanger the public is to prevent them. In a similar way, the best system performance is effected when malfunctions are eliminated. Reliability is the interface between quality assurance and safety. Reliability can neither be tested nor legislated into equipment; it must be built in. High quality in design, procurement, installation, and operation will lead to a system that has high availability, good reliability, and a low probability of incurring an accident. Quality assurance is a total systems approach to achieving these aims. Quality assurance does represent an increase in costs; this increase must be balanced against safer operation and savings resulting from less time lost, fewer repairs, and better control. The prime responsibility for an effective quality assurance program lies with the owner/operator of the plant, who may delegate portions of the program to major suppliers.

5.2.10 Nuclear Energy System

Reactor-System Assembly. To achieve a self-sustaining, but regulated fission process, and for the energy released to be extracted and converted to electricity, a reactor system is constructed.

The nuclear fuel, usually uranium, is fabricated into fuel elements. The typical design for the fuel of a light-water power reactor involves the fuel in oxide form. Where the fuel is uranium, the uranium dioxide is fabricated into pellets, right circular cylinders approximately 19 mm high and 8 mm in diameter.

In light-water reactors, the uranium dioxide material is typically enriched to a low value, approximately 3% to 7%, in the fissionable isotope ^{235}U . This enrichment is necessary because light water has an appreciable neutron-absorption cross section. The extra neutrons available from the added fissile material compensate for the absorption in the moderator. The fuel-pellet material is usually of a ceramic nature (e.g., UO_2); the pellets are dished at both ends to allow for differential thermal expansion and fuel volumetric growth with burnup.

The pellets are inserted into fuel tubes, typically thin-walled tubes of stainless steel or Zircalloy. An open space (with the column of pellets spring-loaded) at the top of the tube is provided to accommodate generation of gases during the fission process. The tubes are sealed top and bottom and are assembled into a configuration involving fixed spacing in a fuel assembly. A representative fuel assembly is shown in Fig. 5-13. This assembly has an overall length of approximately 4.5 m with an active length of approximately 3.8 m.

Plutonium, ^{239}Pu , is produced in the fuel elements during power operation by the absorption of neutrons in the ^{238}U . This material is fissionable and may be recovered during fuel reprocessing and fabricated into new fuel elements. However, to date, regulations in the United States have prevented fuel reprocessing, largely due to concerns about proliferation of materials for use in weapons. Consequently, spent fuel from U.S. plants is currently being stored, awaiting future decisions about reprocessing. In contrast, reprocessing plants have been constructed in Europe.

In gas-cooled reactors in the United States, the fuel-element design differs from that of light-water reactors. The recent gas-cooled reactor elements are hexagonal graphite blocks into which blind longitudinal holes are drilled to receive rods of fuel particles. The fissile material is enriched uranium carbide, UC_2 . Kernels of this material are coated with a pyrolytic carbon-silicon carbide-pyrolytic carbon sandwich. Fertile material in the form of thorium oxide, ThO_2 , kernels is also used. A fertile material is one which, by absorption of neutrons, is changed to a material which can be fissioned. In this case, ^{232}Th is converted to ^{233}U , which has superior characteristics for fission reactions. The kernels are coated with two layers of pyrolytic carbon. The two types of fuel particles are mixed in the proper proportions and are formed with a carbon matrix into fuel "rods" about 15.6 mm in diameter and about 60 mm long. These rods are inserted into the holes in the graphite blocks. Through holes are provided in the block for the helium coolant flow. These loaded graphite blocks or "fuel assemblies" form the basic module for the core of the gas reactor.

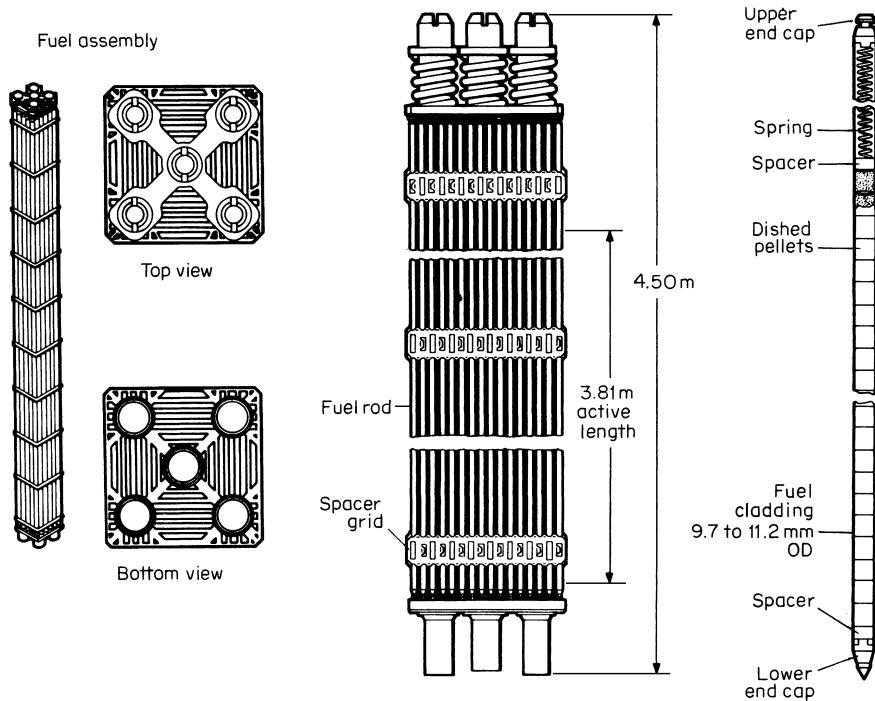


FIGURE 5-13 PWR fuel assembly.

The required number of fuel assemblies to produce a power output desired for the reactor plant are assembled into a reactor-core configuration approximating a right circular cylinder. This configuration provides a high volume-to-surface ratio which minimizes the neutron leakage and conserves the neutrons produced for further fission action. For a 1300-MW (electrical) light-water nuclear plant, a representative core assembly might involve 241 fuel assemblies each weighing approximately 660 kg, for a core equivalent diameter of 3.6 m, a core height of approximately 5 m, and a total core weight of approximately 160 metric tons.

Control-Element Assemblies. In each fuel assembly, several holes are shown. These open holes are spaces into which control elements are inserted for regulation of the fission process. The individual control elements may be grouped typically into control-element assemblies. Control elements for current Pressurized Water Reactors (PWRs) (Fig. 5-22) are located in the fuel elements in this fashion. Control elements of Boiling Water Reactors (BWRs) are blade-type cruciform units. These units are inserted into or withdrawn from the spaces between the fuel assemblies. The control-element assemblies are selected from a material which absorbs neutrons; therefore, by insertion into the fuel assembly or withdrawal from the fuel assembly, the amount of neutrons available for fission production can be reduced or increased, respectively, as required for reactor-system performance. These control-element assemblies are inserted or withdrawn by electromechanical or hydraulic drive mechanisms.

Moderator and Heat-Transfer Medium. The thermal energy released from the core must be conveyed to the electric generator at a rate and in a fashion which meets the requirements. Some consideration has been given to the use of a reactor core to heat gas which is supplied to a magnetohydrodynamic generator. However, commercial systems for the present and near future will continue to use steam turbines as the motive power for the generator.

If fuel of low enrichment (e.g., 3% to 4%) in ^{235}U or other fissile isotope is used, a moderator is needed to take advantage of the larger cross section at thermal neutron energy. If enrichments greater than 20% are used, sufficient fissile material is present to overcome the nonfission capture effects of ^{238}U , and a moderator is not needed. In this case, the reactor is said to be “fast” (referring to the neutron velocity), and liquid metals (selected for low neutron absorption) are used as a coolant.

5.2.11 Plant Arrangement

Primary-System Configuration. The fuel elements, assembled into the core arrangement, are positioned within a reactor vessel by support structures also referred to as reactor internals. The reactor vessel also contains, guides, and directs the primary coolant. Elementary configurations are shown in Fig. 5-14.

In the BWRs, the reactor vessel also contains the steam-separation apparatus, since the coolant is converted to steam in the core. Steam is piped from the reactor vessel to the turbine and condensate

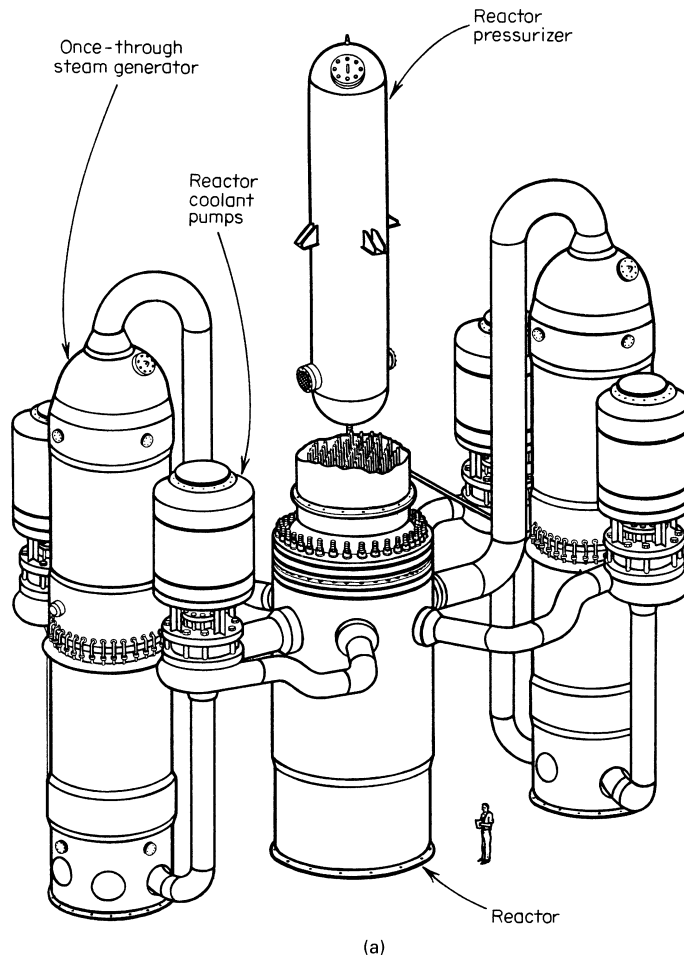


FIGURE 5-14 (a) PWR arrangement with once-through steam generators; (b) BWR arrangement. (General Electric Co.)

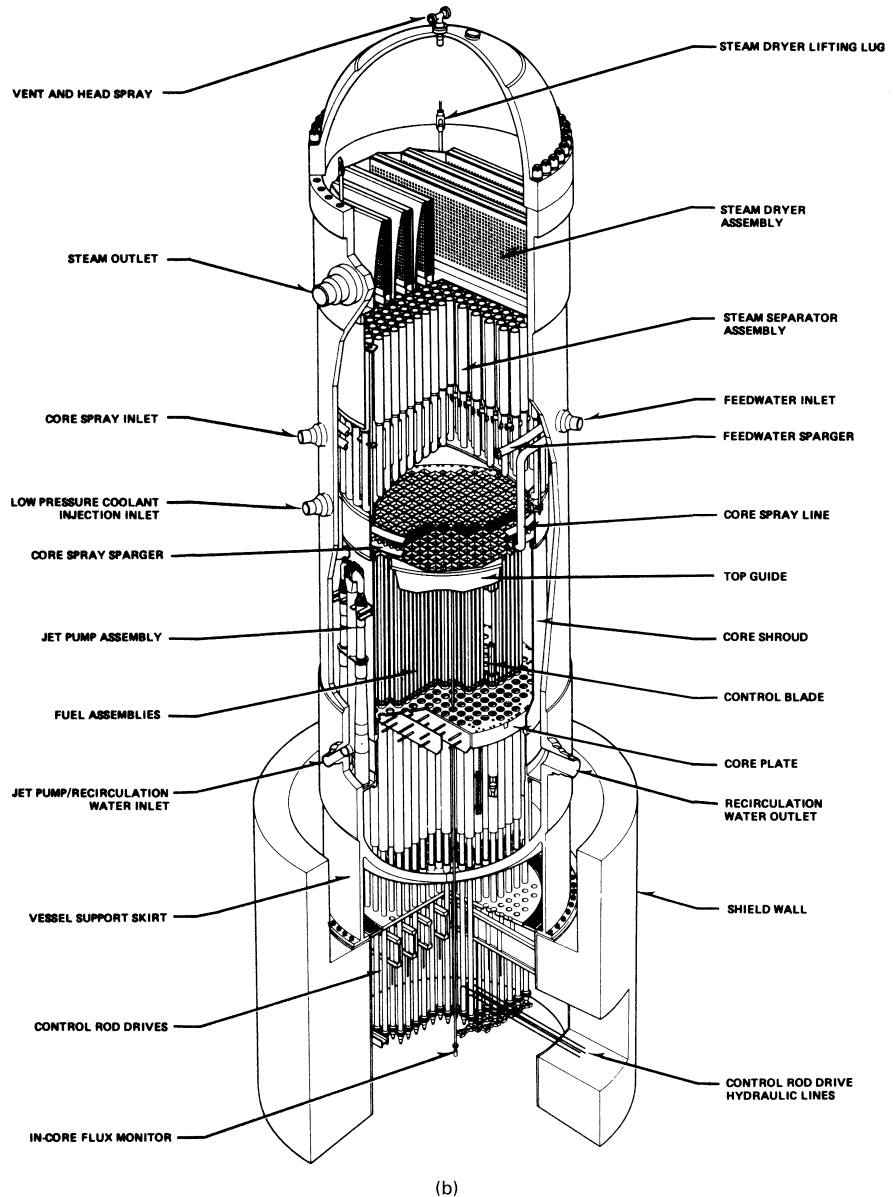


FIGURE 5-14 (Continued)

is returned from the hotwell through feedwater systems to the reactor vessel. Recirculation loops on the reactor vessel increase the recirculation ratio to effect better steam separation and provide better control of void fraction within the core (reactivity control). Access to the core for control rods and for in-core instrumentation is provided from the bottom of the reactor vessel to avoid the steam-separation area at the top.

In PWRs, the primary system is maintained at a subcooled condition by operating at a pressure greater than saturation. Conventionally this pressure is in the range of 12 to 16 MPa. This pressure is maintained by a pressurizer connected to the primary piping. A steam-water interface is maintained in the pressurizer by the action of electric resistance heaters which boil water to raise the pressure or spray flow to quench steam and lower the pressure. The reactor vessel is connected, by heavy piping, to one or more steam generators, and coolant is circulated through this primary system by large pumps. On the secondary side of the steam generator are located the customary steam piping complex, the turbine condenser, and feedwater system.

In liquid-metal reactors, the concerns associated with coolant metal–water reaction in the event of leakage and the induced activity in the primary metal sometimes direct the design to an intermediate heat-exchange system between the primary-metal coolant and the steam system.

Containment Structure. To provide a secondary barrier for radioactivity in the fluid systems and a fourth barrier for the activity in the fuel (the fuel matrix or ceramic, the fuel sheath, and the primary-system boundary are the other three), the primary-system components are located within a containment structure. This structure is designed to confine gases and materials that might occur in the event of an inadvertent release from the primary barrier(s).

The principal types of containments that have been used are

1. The steel sphere made of sections of steel sheet welded together
2. The “inverted lightbulb” of the BWR, which is a concrete cavity in the shape described with a pressure-retaining steel liner
3. The domed cylinder for the PWR; the cylinder of reinforced concrete with an impervious internal steel membrane
4. The cylindrical prestressed-concrete reactor vessel (PCRV) for the GCRs

These structures serve to provide isolation and shielding for the primary-system components. Instrumentation, control, and electric power conductors must pass through the pressure seal while maintaining its integrity. To provide this capability, banks of containment penetrations are provided. The requirements for these items are described by IEEE 317, *Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations*.

5.2.12 Plant Operations

Nuclear Plant Costs. A large amount of equipment and capital investment is required for a nuclear plant. A nuclear steam-supply system (NSSS) is arranged and equipped to initiate, sustain, and regulate the fission process in the reactor fuel, transfer heat from the fuel to the steam generators, and produce steam to supply a turbine. Auxiliary fluid systems to provide chemical cleaning and conditioning of the primary-system fluid, control of chemical shim (if used), supply and purification of secondary water (if required), and processing of radioactive effluents (liquid and gas) are associated with the NSSS. Equipment for instrumentation, control, protection, and electric power distribution is included. Typical equipment supplied in a nuclear steam-supply package is shown in Table 5-4.

This large investment in equipment encourages the development of nuclear plants of large capacity to reduce the per megawatt cost. Plants in the current generation have been from 500 to 1300 MW in electrical capacity. Depending on the size of the system, a single nuclear plant can represent 10% to 20% or more of the system operating capacity. This large size, the high capital investment, plus a low fuel cost of the nuclear plant direct the base loading of the nuclear plant.

Performance Evaluation. Evaluating the nuclear steam-supply system for performance as a source of steam energy often requires that the system be modeled, that is, the system equations be developed. The principal components whose characteristics are important for transient analysis are the reactor fission process and thermal process and the primary coolant piping and the steam generator(s) in a PWR along with the associated control and protection systems.

The functions of these components, in combination, are (1) the reactor core through the fission process converts potential energy in the uranium atoms to thermal energy in the fuel elements; (2) in a

TABLE 5-4 Typical Equipment in a Nuclear Steam-Supply Package

Reactor system:	Rod-drive servicing equipment
Pressure vessel and internals	Vessel-servicing equipment
Control rods	Auxiliary systems, PWR:
Rod-drive mechanisms	Chemical and volume control
Recirculation pumps, BWR only	Residual-heat removal
Steam generators	Spent-fuel pit
Pressurizer	Safety-injection system
Controls and instrumentation:	Component-cooling system
Reactor controls and instrumentation	Radioactive-waste processing
Rod-drive position indicators	Sampling system
In-core instrumentation	Auxiliary systems, BWR:
Ex-core nuclear instrumentation	Reactor-water cleanup
Plant protection system	Standby-liquid control
Plant monitoring and supervisory system	Core-isolation cooling
Steam bypass control equipment	Residual-heat removal
Auxiliary system controls and instrumentation	Core spray system
Control-room panels	High-pressure coolant injection
Feedwater regulating equipment	Fuel-pool cooling and filtering
Tools and servicing equipment	Radioactive-waste processing
Fuel-handling equipment	

PWR, GCR, LMFBR, the primary coolant piping conveys the thermal energy to the steam generator(s) and the steam generator(s) transfer the energy from the primary fluid to the secondary fluid. Since the secondary fluid is maintained at a lower pressure, boiling is introduced and steam is generated; (3) in a BWR, the directly produced steam is separated; (4) the PWR pressurizer maintains pressure on the primary coolant to assure that the primary thermal process takes place in a subcooled condition; and (5) the BWR recirculation system maintains a circulation of primary fluid sufficient to ensure adequate steam separation and to regulate the void fraction in the core (reactivity control).

Neutron Multiplication. The description of performance of the reactor begins with the neutron kinetics, the time behavior of the fission neutrons. A generation of neutrons begins with a neutron flux density from the previous generation (or from the source). A fuel nuclide absorbs a neutron and probably undergoes fission. A small percentage of absorbed neutrons do not cause fission; so the number of neutrons released per capture of a neutron is given by

$$\eta = \nu \frac{\Sigma_f}{\Sigma_u} \quad (5-10)$$

where η = number of neutrons released per capture

ν = number of neutrons released per fission

Σ_f = cross section of the fuel for fission

Σ_u = absorption cross section (fission and nonfission) in the fuel

In a reactor assembly where the fissions are initiated principally by thermal neutrons, some fissioning will be introduced by the fast neutrons before they have been thermalized. The ratio of the total number of fast neutrons produced by neutrons of all energies to the number produced by thermal neutrons is given by ϵ .

During the slowing-down (thermalizing) process, some neutrons are captured in nonfission processes; the fraction escaping such capture is ρ . When the neutrons have been thermalized, they will diffuse in the core region until absorbed in the fuel or in some other material, structure, moderator, poison, etc. The fraction absorbed in the fuel is given by

$$f = \frac{\text{thermal neutrons absorbed in fuel}}{\text{total thermal neutrons absorbed}}$$

If n neutrons are present in one generation (in an infinite system), the multiplication factor, k_{inf} , or the ratio of the neutrons in one generation to those in the next generation, is given by

$$k_{\text{inf}} = \frac{n\eta \epsilon \rho f}{n} = \eta \epsilon \rho f$$

This is also known as the four-factor formula.

In a reactor system of finite dimensions, there is also leakage out of the system, so that the infinite multiplication factor must be adjusted to provide for leakage. Considering the diffusion process and the boundary conditions, it is evident that for a reactor of finite physical dimensions, there will be some leakage (loss) of neutrons from the boundary. In describing the process of slowing down the neutrons, two quantities are developed. The first of these is the diffusion length L , which is equal to one-sixth of the net vector distance that a monoenergetic neutron travels from its source to the point where it is absorbed by a nucleus. A second quantity is the buckling B , which represents the “bending” or appreciable reduction of the value of neutron flux at any point in the reactor.

These quantities may be used to develop two factors which take into account the finite size of the reactor and the leakage which occurs. For the first factor, the term $e^{-B^2 L_s}$ represents the nonleakage probability of the neutrons as they slow down, where L_s is a slowing-down length. The algebraic loss, by diffusion, of thermal neutrons in a volume element is $-D\nabla^2 \phi = DB^2 \phi$. The ratio of thermal leakage to thermal absorption is

$$\frac{\text{Thermal leakage}}{\text{Thermal absorption}} = \frac{DB^2 \phi}{\Sigma_a \phi} = L^2 B^2 \quad \text{so} \quad \frac{D}{\Sigma_a} = L^2 \quad (5-11)$$

Adding the thermal absorption to the thermal leakage, effectively adding unity to both sides of the equation, and inverting, gives, for the second factor, the ratio

$$\frac{\text{Thermal absorption}}{\text{Thermal leakage} + \text{thermal absorption}} = \frac{1}{1 + L^2 B^2}$$

which accounts for the nonleakage probability at thermal energy.

For a finite reactor then, the effective multiplication factor may be expressed by a combination of these two factors:

$$k_{\text{eff}} = k_{\text{inf}} \frac{e^{-B^2 L_s}}{1 + L^2 B^2} \quad (5-12)$$

When fission occurs, more than 99% of the resulting neutrons are produced within 10^{-3} s. The remaining neutrons are produced during the decay of the fission fragments. The time required for their production varies; they may be separated into groups for convenience. These delayed neutrons are essential to the regulation of the fission process.

Reactor Kinetics. For development of control equations, a single delayed group model (Fig. 5-15) may be used for approximation of the neutron production.

The production of neutrons for fission initiation including generation and thermalization is given by $K_{\text{inf}} \Sigma_a \phi e^{-B^2 L_s}$, where ϕ is the neutron population, Σ_a is the absorption cross section, and $e^{-B^2 L_s}$ is the nonleakage probability during thermalization.

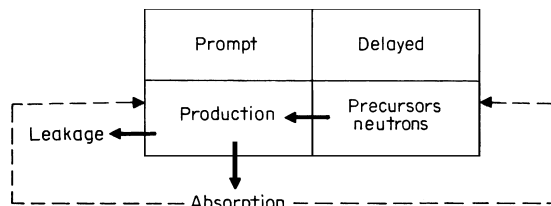


FIGURE 5-15 Single-neutron delayed group model.

The leakage of neutrons is $D\nabla^2\phi = -DB^2\phi$. Assuming β is the fraction of neutrons delayed, the neutron balance for the main group is

$$\dot{n} = (1 - \beta)k_{\text{inf}}\nu\Sigma_{\alpha}\phi e^{-\beta^2 L_s} + \lambda C - DB^2\phi - \Sigma_{\alpha}\phi \quad (5-13)$$

where λ = decay constant of the delayed neutron precursors

C = population of delayed neutron precursors

ϕ = neutron fluence rate (flux)

Since $\phi = n\nu$, the equation becomes

$$\dot{n} = n[(1 - \beta)k_{\text{inf}}\nu\Sigma_{\alpha}e^{-\beta^2 L_s} - \nu DB^2 - \nu\Sigma_{\alpha}] + \lambda C$$

But

$$k_{\text{eff}} = \frac{k_{\text{inf}}e^{-\beta^2 L_s}}{1 + L^2 B^2} \frac{1}{\bar{l}} = \nu\Sigma_{\alpha}(1 + L^2 B^2)$$

and $D = \Sigma_{\alpha}L^2$, where $\bar{l}$ is the average lifetime of the neutrons.

Substituting for k_{inf} and D gives

$$\dot{n} = n[(1 - \beta)k_{\text{eff}}\nu\Sigma_{\alpha}(1 + L^2 B^2) - \nu\Sigma_{\alpha}(1 + L^2 B^2)] + \lambda C$$

Substituting $1/\bar{l}$ for $\nu\Sigma_{\alpha}(1 + L^2 B^2)$ gives

$$\begin{aligned} \dot{n} &= n \left[(1 - \beta) \frac{k_{\text{eff}}}{\bar{l}} \frac{1}{\bar{l}} \right] + \lambda C \\ &= n \frac{(1 - \beta)k_{\text{eff}} - 1}{\bar{l}} + \lambda C \end{aligned}$$

The balance equation for the delayed group is

$$\dot{c} = n(\beta k_{\text{inf}}\nu\Sigma_{\alpha}e^{-\beta^2 L_s}) - \lambda c \quad (5-14)$$

Substituting

$$\frac{k_{\text{inf}}e^{-\beta^2 L_s}}{1 + L^2 B^2} = k_{\text{eff}}$$

gives

$$\dot{c} = n[\beta k_{\text{eff}}\nu\Sigma_{\alpha}(1 + L^2 B^2)] - \lambda c$$

and substituting $1/\bar{l} = \nu\Sigma_{\alpha}(1 + L^2 B^2)$ gives

$$\dot{c} = n \frac{\beta k_{\text{eff}}}{\bar{l}} - \lambda c$$

Rearranging the equation for $\dot{n}$ gives

$$\dot{n} = n \frac{(k_{\text{eff}} - 1) - \beta k_{\text{eff}}}{\bar{l}} - \lambda \dot{c} = n \frac{k_{\text{eff}}(1 - \beta) - 1}{\bar{l}} - \lambda \dot{c}$$

Since k_{eff} is very close to 1, $\beta k_{\text{eff}} \approx \lambda\beta$ and reactivity ρ is the ratio of the excess multiplication factor to the effective multiplication or

$$\rho = \frac{k_{\text{eff}} - 1}{k_{\text{eff}}} = \frac{k_{\text{ex}}}{k_{\text{eff}}} \quad \text{or} \quad \frac{\delta k}{k_{\text{eff}}}$$

Where the deviations from criticality are small, $k_{\text{eff}} \approx 1$ and

$$\rho \approx k_{\text{ex}} = k_{\text{eff}} - 1 = \delta k \quad \text{so } \dot{n} = n[(\delta k - \beta)\bar{l}] - \lambda C$$

The power level P is proportional to the neutron concentration. There are typically six delayed neutron groups. The balance equations then become

$$\dot{P} = \frac{P(\delta k - \beta)}{\bar{l}} + \sum_{j=1}^{\phi} \lambda_j C_j \quad (5-15)$$

$$\dot{C}_j = P\beta_j\bar{l} - \lambda_j C_j \quad (5-16)$$

where j represents the delayed neutron group and β_j , λ_j , C_j are the fraction, decay constant, and concentration of delayed neutrons, respectively, of the j th group.

The balance equations show that, in the steady state, the effective multiplication factor is equal to 1. If the K_{eff} increases above 1, the multiplication will increase with time; if K_{eff} decreases below 1, the multiplication will decrease below 1. If $k_{\text{eff}}(1 - \beta) - 1$ increases to a value of 1 or greater, the reactor is said to be prompt critical and the rate of power increase depends on the ratio of $k_{\text{eff}}(1 - \beta) - 1$ to l . Since l is so small (about 10^{-4} s or less for a thermal reactor; 10^{-7} s or less for a fast reactor), P increases very rapidly with time for any appreciable value of $k_{\text{eff}}(1 - \beta) - 1$. Regulation of the process at these rates with conventional apparatus is very difficult. For this reason, k_{eff} in power reactors is kept below the value $1/(1 - \beta)$ when the reactor is operating.

Reactivity Control. The reactivity is affected by neutron absorption. The absorption occurs principally from control-element-type absorbers, dissolved-chemical control absorbers, resonance absorption in the fuel, absorption in the moderator, and absorption by fission products. The absorption initiated by the control elements and the chemical shim are varied by the operators and are characterized by δk_c .

The reactivity effect from the fuel is caused by the widening of the resonance peaks (Fig. 5-16) with temperature, which increases the nonfission capture of neutrons. With cores containing large amounts of ^{238}U and ^{232}Th , this Doppler effect is negative; that is, increasing the power level introduces a reactivity change which opposes the increase. The Doppler coefficient varies with coolant and fuel temperature and with moderator voids. Typical values of the Doppler coefficient are shown in Fig. 5-16. The curves may be approximated by the equation

$$\delta k_F = AT_F - BT_F^2 \quad (5-17)$$

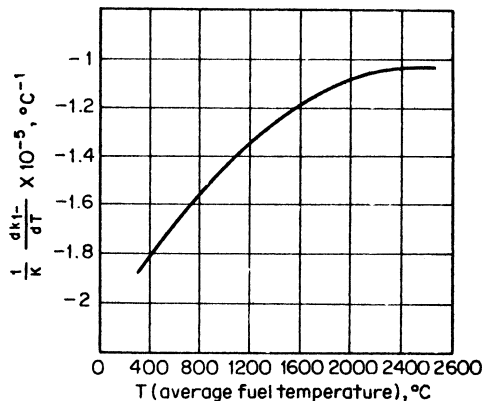


FIGURE 5-16 Values of the Doppler coefficient of reactivity. (General Electric Co.)

The reactivity effect of the moderator depends on the type of moderator and the type of reactor. In water-cooled and moderated reactors, the reactivity effects are initiated by changes in density which affect the slowing-down power and the absorption. Boiling-water reactors are operated at a relatively constant pressure and saturated conditions, which correspond to a relatively constant temperature. Density changes due to temperature are small. The steam production varies with power level so that density variation by voids is appreciable. In a pressurized-water reactor, the coolant in the core is subcooled and voids are suppressed. Coolant temperature may be varied with power, which would cause density changes as a function of temperature.

A reactivity change from the moderator based on the density change also occurs if

there is dissolved neutron absorber (chemical shim) in the moderator. A density decrease causes less of the absorber to be present in a given volume and a decrease in neutron absorption with increasing temperature.

The reactivity effect of voids and of temperature change of the water, therefore, is to oppose a change in power while the reactivity effect, due to temperature change, of dissolved chemical shim is to aid a change in power. Since it is desirable (for safety reasons) to have a negative moderator temperature coefficient of reactivity, that is, a coefficient that with decreasing moderator density acts to retard the fission process, the amount of dissolved chemical shim is usually limited to that which will do no more than reduce the temperature coefficient of reactivity to zero.

Another reactivity effect is produced by the buildup of fission products. Some of these are neutron absorbers and will act as a retardant of the fission process by removing active neutrons. Two of the strongest absorbers are xenon, ^{135}Xe , and samarium, ^{149}Sm , whose absorption cross sections are 3×10^6 barns and 5×10^4 barns, respectively. ^{135}Xe is produced directly as a fission fragment (fission yield 0.3%), and in the decay chain of the fission fragment ^{135}Te (fission yield 5.6%) is



On neutron absorption, ^{135}Xe is converted to ^{136}Xe , which is stable and has a low neutron cross section. ^{135}Xe , therefore, has two modes of production and two modes of elimination. This can be represented by the equation

$$\begin{aligned} \frac{dX}{dt} &= (\lambda_x \Sigma_f - \sigma_x X) \phi + \lambda_1 I - \lambda_x X \\ \frac{dI}{dt} &= \gamma_1 \Sigma_f \phi - \lambda_1 I \end{aligned} \quad (5-18)$$

where X = number of atoms of ^{135}Xe present per cubic centimeter (cm^3) at any time t

γ_x = fractional yield of xenon as a direct fission product

σ_x = microscopic thermal-neutron absorption cross section of ^{135}Xe

ϕ = thermal neutron flux

λ_1 = decay constant of ^{135}I

I = number of atoms of ^{135}I present per cm^3 at any time t

λ_x = decay constant of ^{135}Xe

Σ_f = macroscopic fission cross section of fuel in reactor

γ_1 = fractional yield of ^{135}I from direct fission process

The concentration reaches an equilibrium value during steady-state operation of the reactor but undergoes transients as the power changes. Of special concern to reactor regulation is the variation that occurs when the core is made subcritical following a power history. With the resulting large decrease in Xe removal by neutron absorption, the concentration of Xe increases because the difference in the ^{135}I decay and the ^{135}Xe decay allows a buildup. The peak concentration of ^{135}Xe is proportional to the preshutdown power level, as shown in Fig. 5-17. This absorbing effect must be overridden by control rods or elements if start-up is to occur.

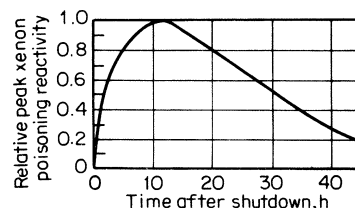


FIGURE 5-17 Relative peak xenon-poisoning reactivity.

Thermal Reaction. The heat from the core is generated in the fuel material as a result of the fissions initiated by the impinging neutrons. Although the center of a fuel element is subject to self-shielding, the fuel-element radial temperature distribution can be obtained by assuming a uniform volumetric heat source in a conduction problem. The gradient from the centerline of the fuel element through the gas gap and cladding to the coolant might be as shown in Fig. 5-18a. The power distributions (Fig. 5-18b and c) axially and radially in the core also vary, generally being highest in the center and decreasing toward the top and bottom and outside.

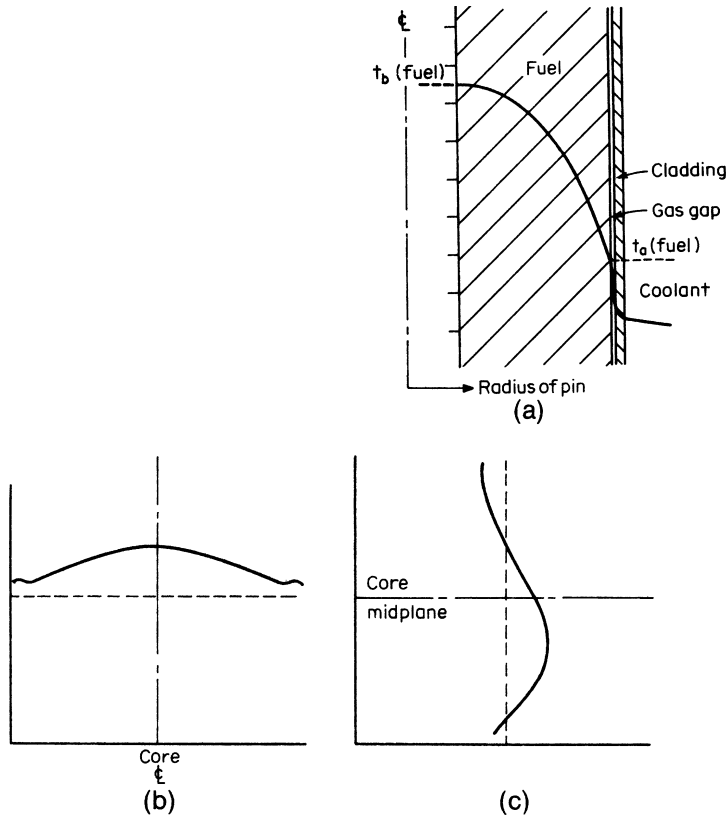


FIGURE 5-18 (a) Temperature distribution in fuel rod; (b) radial power profile; (c) axial power profile.

Control elements are adjusted to provide some degree of flattening. This makes a higher average power possible from the reactor for a given peak power.

For instrumentation and control purposes, the fuel may be considered to be mathematically represented by a series of time lags. These lags are important because the protection and control are strongly dependent on the time involved in transferring the heat out of the fuel. Since the elements are very long compared with their radius, conduction of heat in the axial direction can be neglected. In cylindrical coordinates, the Laplace equation for heat flow (neglecting the Z direction) is

$$\frac{\partial^2 T}{\partial \gamma^2} + \frac{1}{\gamma} \frac{\partial T}{\partial \gamma} + \frac{1}{\gamma^2} \frac{\partial^2 T}{\partial \phi^2} = \frac{\rho C}{k} \frac{\partial T}{\partial t} \quad (5-19)$$

where T = temperature, $^{\circ}\text{C}$

t = time, s

k = thermal conductivity, $\text{W}/(\text{m})(^{\circ}\text{C})$

C = specific heat, $\text{J}/(\text{kg})(^{\circ}\text{C})$

ρ = density, kg/m^3

γ and ϕ = cylindrical coordinates, m

Solving this equation and assuming that the heat capacity of the cladding and the thermal resistance from the surface of the fuel pellet to the cooling channel can be neglected, gives the fuel-element transfer function

$$G(s) = (1 - \gamma) \sum_{n=1}^4 \frac{F_n}{1 + \tau_n s} + \gamma \quad (5-20)$$

where $G(s)$ = fuel transfer function

γ = fraction of heat produced by photons

F_n = gain of n th term

τ_n = time constant of the n th term

Application of Performance Equations. The equations of performance for the various portions of the NSSS may be used for mathematical analyses or for development of simulations. Simulation of an NSSS is frequently desired. Such simulation enables (1) evaluation of system performance to assure that plant performance requirements are met, (2) performance evaluation of monitoring or control equipment, and (3) analyses of protection action to assure protective-system adequacy.

The simulation may be performed with an analog, a digital, or a hybrid computer; selection of the appropriate method should be based on the equipment to be simulated and the objectives of the tests to be performed.

The level of detail of the simulation also varies with the objectives of the test. For instance, modeling of the reactor core may be one node for gross thermal input to another component or be multinode to evaluate the performance of in-core instrumentation; 1-, 4-, 7-, and 38-node models are examples of core simulations that have been found to be useful.

Since commercial power-reactor systems are large and are oriented to power production, it is inconvenient to encumber them with apparatus and operations directed to obtaining their time-response characteristics, which interferes with normal operation. It has been found that the discrete nature of neutrons and the statistical nature of the fission process give rise to random fluctuations in neutron population or "reactor noise." The production, absorption, and leakage of neutrons can be considered analogous to the random flow, from emitter to collector, of electrons in a diode. Using power-spectral-density techniques, the reactor transfer function can be developed from an analysis of reactor noise.

5.2.13 Control Systems

Fission Regulation. The fission process is regulated by the absorption, in a controlled manner, of some of the neutrons which cause fission. The controlled absorption may be provided by the control rods or control elements which are mechanically inserted into, or withdrawn from, the reactor core or by absorber material dissolved within the reactor coolant. Because steam is produced directly from the coolant in a BWR, dissolved poisons are usually not used in normal operation of the BWR. In a gas-cooled reactor where the coolant gas is either helium or carbon dioxide, it is not feasible to provide dissolved absorbers in the coolant. Dissolved poisons, therefore, are used principally in PWRs. Where dissolved poisons or fixed poisons within the core are used, they are generally used for the purpose of accommodating core burnup. Changes in the reactivity for normal operation, initial start-up, planned shutdown, and restart are normally accomplished by control rods or control elements. Since the rate of change available from dissolved poisons or fixed burnable poisons is normally quite slow, the mechanically inserted control rods or control elements are also used for emergency shutdown.

Considerations involved in determining the characteristics of the control rods or elements are the amount of reactivity that has to be controlled, the position accuracy (which corresponds to the minimum increment of the reactivity) to be provided, the rate of reactivity that must be provided for operation, and the reliability of components. The reactivity requirements of a nuclear system are based on the planned rate of fuel depletion, the fission-product buildup, including the principal poisons xenon and samarium, inherent reactivity effects such as temperature or void changes, and the control range necessary for maneuvering to which the plant may be subjected.

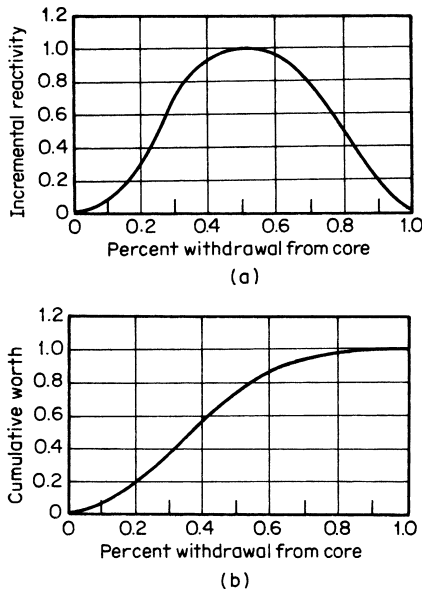


FIGURE 5-19 (a) Incremental reactivity versus core withdrawal; (b) cumulative worth versus core withdrawal.

position of the control rods or elements is an important information in determining core power distribution and it represents a knowledge of the rate at which the fission process is proceeding, it is desirable to provide indication of the position of the control elements and to provide input from the rod-position sensors to core-power-distribution calculations. Position indication can be provided by analog meters, digital indicators, analog presentations on a cathode-ray tube, position logged by a digital computer with printout, or other types of display. Because of the importance of the position of the control element in reactor regulation and reactor shutdown, two systems of indication for the control elements are normally provided.

Controlling Fluid Processes. The primary system of most U.S. reactors involves either light water or helium as a coolant. Other fluid systems are provided to supply makeup, effect cleanup, process waste, etc. Conventional instrumentation is used to monitor these variables, and control signals in accordance with preselected control program are applied to appropriate actuator elements.

The special requirements of nuclear systems for continuity of cooling may require that extra care be used in the application of control equipment to assure high reliability. Establishment of set points and alarm points should also be done with due care.

Protection Systems. The high specific power of nuclear reactor sources coupled with the potential for the release of radioactivity, which might be a hazard to human beings, requires that additional systems be provided for regulation. In addition to the instrumentation provided for the control of the fission and fluid processes, systems are specifically supplied to initiate protective action in the event that preselected limits are exceeded. The relation of protection systems to other instrumentation and control systems is shown in Fig. 5-20.

Basically, a protection system must provide a functional capability to initiate action in the event of a design-basis event which, if unchecked, could lead to unacceptable consequences. Because of this requirement, the protection system must operate when required and must operate correctly. The development of functional and reliability requirements is very important in this arrangement.

The control rods or elements are also used to flatten the power developed radially and axially within the reactor to raise the average power and increase the output from the system. To accomplish this purpose, the rods or elements are normally assigned to groups or banks which are operated together. The capability of a rod or element to perform its neutron absorption or the worth of the control unit is dependent on the position of the element within the core. As the element progresses from the bottom, its worth increases from a low value to a peak and then decreases again to a low value as it is withdrawn from the top of the core. The typical incremental worth and the cumulative worth of an assembly are shown in Fig. 5-19. (Individual worths are affected by core power distribution.)

Emergency Shutdown. In order to provide an emergency shutdown capability, the control rods or elements are normally provided with a fast-insertion capability. This characteristic, also referred to as scram, is provided for control drives mounted on top of the reactor by a detaching capability with the rods or elements free-falling into the reactor core. With rods or elements that are mounted on the bottom of the reactor, a capability is provided to drive the rods rapidly up to their full insertion position. The latter mechanism is typically hydraulically actuated. Since the

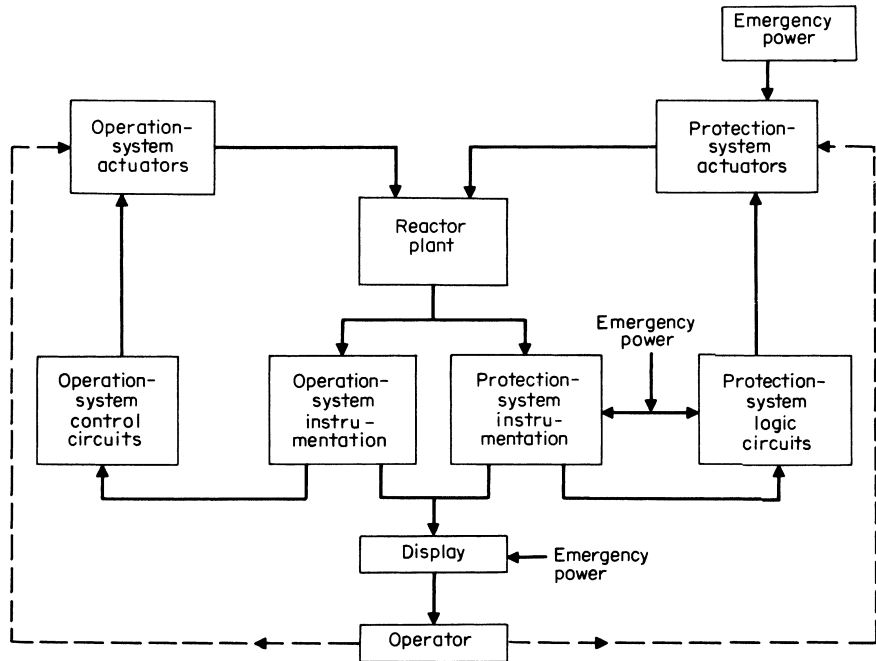


FIGURE 5-20 Relation of protection system to other instrumentation and control system.

The consequence of the greatest concern associated with a nuclear system is the release of radioactivity. The nuclear fuel itself has multiple barriers between the fuel material and the outside boundaries for the public. These barriers are the fuel matrix, the fuel jacket or cladding, the primary system, and the containment structure. For any given condition, it is the duty of the protection systems to prevent a situation which would lead to an excessive release of radioactivity from occurring past a given boundary.

The protection systems therefore include the reactor shutdown system and other systems that effect the containment of the radioactivity such as emergency core cooling, containment isolation, containment pressure reduction, emergency power sources, and air filtration. A protection system itself includes the instruments, logic systems, actuators, protective interlocks, and mechanisms which carry out the necessary functions. That system which effects reactor shutdown, normally referred to as the reactor protection system, includes all electrical and mechanical devices and circuits used to initiate a reduction of the fission process below criticality. The engineered safety features or engineered safeguards systems include everything else associated with protection except the reactor protection system.

In order to accomplish a high availability, that is, the capability to act when needed, it is necessary to provide multiple channels to perform the same action. The probability of any one channel not being able to act at a given time, therefore, is offset by having other channels capable of performing the action. In order to avoid spurious action, that is, initiation of a protection when none is required, it is usually the practice to provide a logic arrangement and initiate action only when there is a coincidence of two or more channels. The consequences of spurious action, for example, loss of a power source, in a large system can also be rather severe in terms of impact not only at the plant site but also in the area of public usage of the plant output, and such false action should therefore be precluded. Conventional protection systems have used a logic arrangement involving three or four channels and requiring the coincidence of two of three or two of four in order to initiate action.

In the selection of the plant variables that are sensed by the protection system, it is usual to evaluate those plant conditions which could occur as a result of some event. In some cases, such as those involved

with protection of the clad integrity, the variables cannot be measured directly and an inferred or computed variable must be used to initiate the protective action. Such plant variables used to evaluate the fuel design limits are the departure from nucleate boiling (DNB) ratio in the case of PWRs and minimum critical heat-flux ratio (MCHFR) in the case of BWRs. In order to maintain these critical values below safety limits, it is necessary to monitor the observable parameters which affect DNB or MCHFR such as thermal power, coolant flow, coolant temperature, coolant pressure, and core power distribution (from the nuclear instrumentation and control-element-position sensors). These values are translated into systems relating to desired protection, and reactor shutdown is initiated if the measured variables approach boundaries of regions established by these systems. The earlier nuclear plants utilized parameters directly and the shutdown values or limits were set on the basis of calculated values relating to a set of curves. Current nuclear plants involve, to a certain degree, the use of online digital calculators permitting the protection system to provide continual computations of the relation of the variables. Shutdown limits are initiated as a function of the instantaneous value of the measured variables.

In order to avoid the possibility of disabling of the protection system by some common initiating mechanism, a principle of diversity of sensing and operation is suggested. The diversity relates to a different type of equipment or a different mode of operation to effect protection action from a given condition. Types of diversity that have been considered include equipment, functional, operational administrative, and design administrative diversity. Therefore, an evaluation should be made of the utilization of diversity in order to assure that (1) a definite objective may be obtained, (2) the additional complexity introduced by added equipment will not result in a degradation of the system, and (3) typical relations will be maintained between the primary action and the diverse action provided.

Nuclear Instrumentation. Since the fission process involves a neutron fluence where the fluence rate (flux) is proportional to power, it is essential to measure the fluence rate. Such measurement, for the large-core commercial reactors, involves special consideration, including the following:

Measurement over 10 to 13 orders of magnitude may be required.

There may be important spatial variations.

The measurement is of uncharged nuclear particles (neutrons).

The measurements have to be made in a background of substantial gamma radiation.

In addition to the monitoring of neutrons, a nuclear steam system involves the monitoring of radiation, principally gamma, from process lines and fuel.

Ex-Core Neutron Monitoring. Ex-core, or out-of-core, detectors are those which are located external to the reactor core and usually external to the primary pressure boundary. The fast-neutron flux leaking from the core provides a neutron flux spectrum for some distance beyond the vessel wall. This flux, proportional to a spatially average core power, becomes thermalized in the shielding, usually hydrogenous material surrounding the reactor.

The environment in which the detectors are located involves, typically, neutron fluxes from up to 10^{11} neutrons/(cm²)(s), gamma dose rate up to 10^7 R/h, and temperatures to 200°C.

The detectors used are devices which produce a current pulse when subjected to the passage of a nuclear emission. The incident radiation drops some or all of its energy within the detector, causing ions to be produced. The ions produced are attracted to electrodes, within the detector, by the effect of voltage across the electrodes. The number of ions collected is a function of applied voltage. The chambers are filled with a gas selected to enhance the performance for a particular type of operation. The chambers are operated with voltages between the electrodes to effect the collection of charged particles as pulses or current (continuous pulses). Operation for ex-core ion-chamber detectors is normally in the flat or plateau region. The “knee” at the left of the curve moves to the right with increasing ambient radiation, so the selected voltage operating point must be sufficiently high so as to remain on the plateau for all conditions.

The chambers can be made sensitive to neutrons by coating the electrodes with a film of material containing an element with which neutrons interact, such as ²³⁵U or ¹⁰B. Enrichment of the isotope can be selected to effect desired performance in the range of neutron flux. In addition to pulses from

incident neutrons, there are pulses from other incident ionizing radiation such as gamma rays. The contribution from gamma rays can be countered or compensated for by supplying two identical volumes, one with neutron-sensitive coating and one without, and subtracting their output. Since the neutron fluence rate and the gamma level do not increase together, that is, maintain the same ratio, this compensation can be completely canceled only at a given power level. Over the range of operation, about 97% to 98% compensation can be effected. The compensated ion chambers (CIC) can be used in a fluence rate about two decades below that of the uncompensated ion chambers (UIC). Ion chambers are applied for both in-core and ex-core monitoring.

Low neutron fluence rates produce currents too low to be accurately measured with an ion chamber. Proportional counters operating in a pulse-counting mode are conventionally used in this application. These detectors are filled with a gas, such as BF_3 , which interacts with incident neutrons to produce ionized particles. Gas amplification is used to increase the output for a given event. Voltage applied is typically in the center of the proportional range. A well-regulated voltage supply is necessary.

Ion chambers using a sensitive coating involving ^{235}U absorb neutrons and undergo fission which generates the ions. Because of the substantial energy imparted to the ions by the fission process, these detectors are used satisfactorily for both current and pulse generation. Operation over 10 decades of neutron fluence rate may be satisfactorily achieved.

Reliability. Selection and procurement of the equipment must include consideration of minimum maintenance and low-failure-rate characteristics. The steps taken to achieve quality levels include good design practices, quality control, qualification testing, calibration, and system testing.

Independence. The equipment is to be installed so that independence of redundant channels is preserved. This requires that (1) components and circuits be electrically separated to prevent the propagation of electrical faults, (2) components and circuits be physically protected from destructive factors such as missiles and water or steam jets, and (3) steps be taken to avoid loss of protective action in the event of common-mode events such as fire or high temperature.

Signal Validation. The equipment design and arrangement is to be such that there are means of verifying that the signal represents the actual condition of the variable monitored. Such verification may include

1. Calibration
2. Cross checking between channels
3. Introducing and measuring known perturbation in the variable

Maintenance. In order to assure high system availability, redundant parts of the systems must be both repairable and adjustable. Special consideration must be given to access, bypassing, removal of modules, and calibration.

Information Readout. The system readouts are to be designed to provide operators with accurate, complete, and timely information. Consideration must be given to sequence and trend indication and to indication of related conditions.

Emergency Power Systems. Because of the need for protective action to be available at all times when the reactor is operating and the need for continued cooling and monitoring when the reactor is shut down, systems must be provided to assure high availability of electric power.

Primary Coolant Circulators. The largest single plant load is the drives for primary coolant circulation. Since it is important to maintain coolant circulation and since these drives are generally too large to be supplied by engine-driven sources, provisions should be made to supply the coolant circulator drives from two or more sources. Frequently, arrangements are made for the main generator to supply two or more power lines. Provisions in the switchyard enable the plant distribution system to be supplied from the plant generator or from one or more of the outside lines.

In spite of possible connection of plant loads to multiple external power sources, it is possible to lose all external lines, for instance, by a tornado. In this event, a local source of power to supply critical ac loads is required. For these purposes, engine (diesel)-driven generators are usually used. Credit can sometimes be taken for local hydro generators or gas-turbine generators if these sources can meet the requirements. These power systems must be designed so that they provide power to the station following a design-basis event. An ac power system (generation and distribution), a dc power

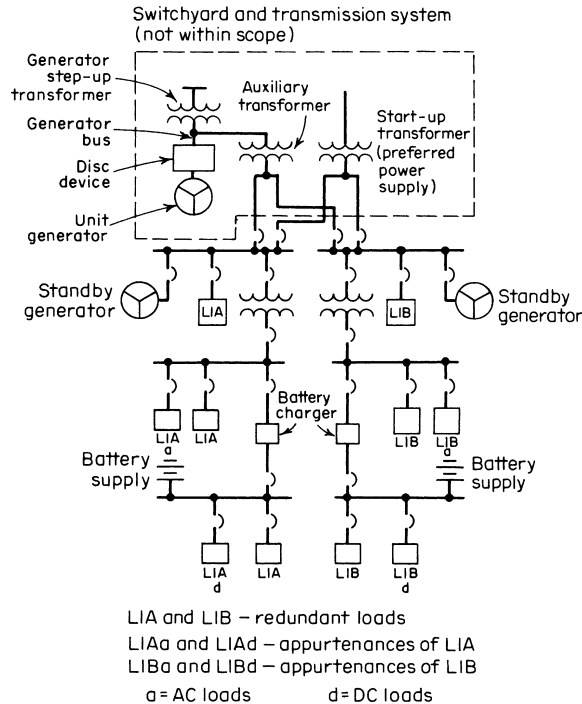


FIGURE 5-21 Class 1E power system for single unit.

system, and a vital instrumentation and control power system are provided. An example of a safety-grade power system is shown in Fig. 5-21.

In the ac system, each of the redundant load groups must have access to both a preferred and a standby power supply. The units of the standby supply must have sufficient independence from the preferred supply and from one another to preclude a common failure mode. Load assignment must be such that the safety actions of each group are redundant and independent. Protective devices must be provided to limit the degradation of the system and maintain the power quality (voltage and frequency) within acceptable limits. Following a demand for the standby power supply, it must be available within a time consistent with the requirements of the engineered safeguards features and the shutdown systems.

In the dc system, batteries, distribution equipment, and load groups are arranged to supply critical dc loads and switching and control power. Redundant load groups, and corresponding battery sections, must be sufficiently independent to preclude common failure modes. Each of the redundant load groups must have access to one or more battery chargers; the batteries are to be kept charged. The battery supplies must be sized to be able to start and operate their assigned loads in the expected loading sequence for a length of time commensurate with the protection provided.

Battery chargers supplying the redundant load groups must have sufficient capacity to restore the battery from its design minimum charge to its fully charged state while supplying normal and postaccident loads. Each charger supply must have a disconnecting device in its ac feeder and one in its dc output line. The dc system must be equipped with surveillance equipment to monitor its status and to indicate actions.

The vital instrument system is provided to power the instrumentation needed for reactor protection and engineered safety features. Since there may be considerable variation in the instrumentation in various plants, the vital system may be required to supply ac or dc or both. To preserve freedom from common-mode failure, the vital supply must be divided into redundant and independent systems with adequate status indication. Provisions for testing, adjustment, and repair should be included in the parts of the emergency power systems to improve reliability and availability.

5.2.14 Radioactive Waste Disposal

There are two broad categories of radioactive wastes, low-level wastes (LLWs) and high-level wastes (HLWs). Nuclear power plants produce significant quantities of both. (Other important sources include defense [military] labs and various commercial/university/hospital labs using radioisotopes, research reactors, etc.) Their safe and economic handling represents a key objective if nuclear power is to remain competitive in the future. Issues related to radioactive wastes are exceedingly complex, involving both intense political-societal-legal interactions and technology. Here we will briefly discuss various technical issues, but caution the reader that sociopolitical factors are, in many cases to date, overriding.

The principal Nuclear Regulatory Commission (NRC) regulation for LLWs is 10 CFR 61. It defines three classes of radioactive waste—A, B, and C—depending both on the isotope's half-life and on the specific activity in curies per cubic meter, class A representing the shortest half-life and less specific activity and class C being the highest. Ideally, the classification would be hazard-based, but for practical reasons and ease of enforcement, it is based largely on the process of the industry that generated the waste. 10 CFR 61 provides a quantitative evaluation for all radionuclides involved. All power-reactor fuel elements are automatically classified as HLWs while resin, sludges, contaminated clothing, materials, etc. are LLWs.

An LLW disposal facility can be licensed either by a state (if qualified by the NRC) or by the NRC itself for use by a commercial operator. The site is selected from appropriate candidate land when extensive facts about the geology, water flow patterns, and nearby population are known or developed. After 20 to 50 years, the site is closed, and ownership is transferred to the state or federal agency, which then continues to monitor the site for the period of institutional control (100 years). Then the license is terminated and no further maintenance is needed, but the design should have assured protection for a period of 500 years.

The disposal of HLWs is controlled by the Nuclear Regulatory Commission, following regulations 10 CFR 60. Some important provisions in this lengthy regulation are as follows:

The facility should not pose an unreasonable risk to the health and safety of the public; the allowed radiation dose limit is a small fraction of that from natural background.

A multiple barrier approach is to be used, including the waste form, the containers, and the site geology.

A thorough site characterization is required with features such as great geologic stability and slow water movement regarded as favorable.

The location should not have attractive resources, should be far from population centers, and under federal or state control.

HLWs are to be retrievable for up to 50 years.

Packaged wastes should be secure for at least 300 years; groundwater travel time to any source of public water should be at least 1000 years.

The annual release of radionuclides must be less than $10^{-3}\%$ (in radioactivity) present 1000 years after the repository is closed.

Predictions of safety must be made with conservative assumptions and by calculations that take account of uncertainties, using expert opinion.

In principle, prior to storage/disposal of radioactive isotopes, power reactor fuel elements could be reprocessed (e.g., by chemical solution followed by separation) to recycle fissile and some structural material so that only the radioactive fission products and activated substances are sent to disposal. However, by presidential decision, based on a concern that reprocessing might enable nuclear proliferation, the United States has not permitted reprocessing. (Other countries, e.g., France, have reprocessing plants.) The only actual reprocessing of commercial wastes done was by Nuclear Fuel Services, Inc., at West Valley, New York, in the period 1966 to 1972. This plant was shut down because it was uneconomical to operate, compared to the cost of producing fuel from uranium ore. Thus, since 1972, spent fuel has been accumulating at nuclear power plants.

Another distinction among radioactive wastes involves transuranic elements. Transuranic (TRU) wastes are those containing isotopes above uranium in the periodic table of chemical elements. They

are the by-products of fuel assembly and weapons fabrication and of reprocessing operations. Their radioactivity level generally is low, but since they contain several long-lived isotopes, they must be managed separately. This classification includes isotopes with half-lives greater than 20 years with a total activity greater than 100 nanocuries per gram of waste material. Isotopes include plutonium-239, americium-241, americium-243, curium-244, neptunium-237, and curium-245.

Transuranic wastes give off very little heat, and most of them can be handled by ordinary methods not requiring remote control. Since 1970, they have been placed in retrievable storage.

Another special category of wastes involved with the power reactor fuel cycle consists of mill tailings. These wastes are the residue from the physical and chemical processing of uranium ore to obtain uranium. They are stored and disposed of separately due to the enormous volume of material involved.

Another waste category must be considered along with "pure" radioactive materials. "Mixed wastes" are those containing both hazardous chemicals and radioactive substances. The disposal of hazardous wastes is regulated by the Environmental Protection Agency (EPA), while radioactive wastes are controlled by the NRC. Thus, it has been necessary to establish consistent dual rules by agreement between agencies. This overlap of agency control results in many complications, which have caused difficulty in handling of mixed wastes, and has led to most such waste being chemically processed to eliminate the chemical hazard, followed by disposal as LLRW.

What to do with spent fuel has been an important challenge for the United States since the early 1970s, and will continue to be well into the twenty-first century. It is estimated that about 80,000 metric tons of spent fuel elements will have to be stored by the year 2020, assuming, of course, that no reprocessing is done.

Several materials have been candidates for the host geologic medium for a disposal site. Regions containing deposits of rock salt, basalt, granite, tuff, and argillaceous materials (clay and shale) have been involved in several levels of investigation. The Yucca Mountain region in Nevada has been selected for detailed evaluation. A facility located deep beneath the surface of the tuff-type mountainous range would be employed. Tuff is a compacted and hardened ash that has come from volcanoes. Yucca Mountain is composed of a welded tuff, formed by a flow of ash into sheets, that has become solid through pressure from material above. The Yucca Mountain site would use the mined cavity method for storage (see the general concept illustrated in Fig. 5-22). A shaft would extend from the earth's surface down to a series of horizontal tunnels. Canisters containing spent fuel would be placed in holes drilled in the tunnel floor. Then the openings would be backfilled. In addition to the geological safety achieved, this method has the advantage of using conventional mining techniques. After a lengthy legal battle with the state of Nevada concerning the government's right to characterize the Yucca Mountain site, plus delays due to various citizens' protests about issues such as possible water seepage into the mine structure, the DOE recently resumed work to characterize and evaluate the site.

A large volume of wastes classified as high-level defense waste, including some mixed with toxic chemical liquids, is now stored in underground tanks and bins at three main government sites—Hanford, Idaho Falls, and Savannah River.

We see from the foregoing that there is one radioactive waste problem but three key disposal challenges: defense wastes; spent fuel; and low-level wastes.

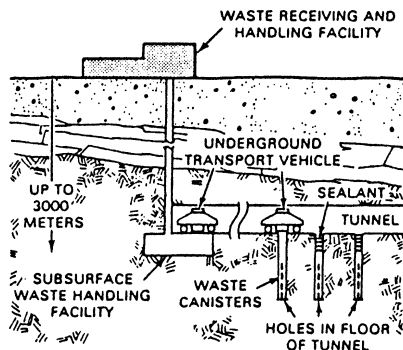


FIGURE 5-22 Emplacement of waste canisters in a mined cavity.

Modern reactors use uranium that has a higher percentage of ^{235}U (3%) than is found in nature (0.7%).

The enriched fuel is typically in the form of uranium dioxide (UO_2) as small pellets about $3/16$ in in diameter and $3/8$ in long.

During the $1\frac{1}{2}$ to 3 years that a fuel element typically stays in the reactor, neutron irradiation has consumed some of the uranium and produced some new materials. The content of ^{235}U is reduced to about 1%, while ^{238}U has gone down from 97% to 94%. A new fissile isotope, ^{239}Pu , has been produced. It has a half-life of around 24,000 years, emitting an alpha particle of 5.1-MeV energy. As noted earlier, a small quantity of other transuranic elements are also produced. By successive neutron absorption ^{239}Pu becomes ^{240}Pu , fissile ^{241}Pu , ^{242}Pu , and

short-lived ^{243}Pu . (Light-water reactors are sometimes called “converters” because they transform some of the ^{238}U into plutonium isotopes.) Figure 5-23 shows the before and after composition of spent fuel. Note that the spent fuel still has most of its original ^{238}U and a fairly high fissile fuel content.

The volume of a typical assembly is less than 7 ft^3 with about 60 assemblies removed per year from each reactor, a total of around 400 ft^3 of spent fuel must be handled. From the 100 reactors in the United States, the annual spent fuel volume would thus be around $40,000\text{ ft}^3$, corresponding to less than 1-ft depth in a standard football field.

The actual amount of radioactive materials in the spent fuel is considerably smaller than the total volume quoted above. If the fuel were reprocessed, these would be extracted, as would most of the plutonium isotopes. The uranium would be cleaned up in preparation for reuse. For each 1200-lb assembly, there would be only around 35 lb of fission product waste. For 60 assemblies discharged per reactor per year, this is about 1 ton. The actual volume would be only 3.4 ft^3 , which is 18 in on a side. However, most wastes could not be stored or disposed of in such a concentrated form because its radioactivity would produce intense heat.

If, as at present, fuel is not reprocessed, two choices are open: store it indefinitely or dispose of it by burial or other isolation technique. These options do not seem optimum in view of the value of the fissile elements in the fuel elements and the reduction in volume that reprocessing could achieve. The decision to avoid storage was based on the Pu-239 content and nuclear proliferation concerns. However, trade-offs among options must be constantly reevaluated with changes in both technology and international politics.

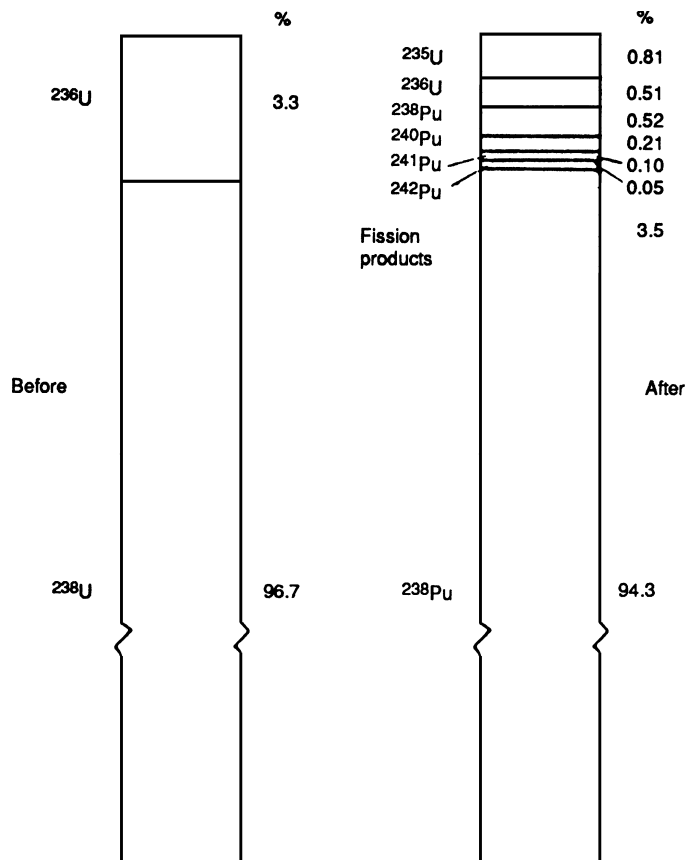


FIGURE 5-23 Composition (in percent) of fresh fuel and spent fuel. Uranium-235 is burned to form fission products and ^{236}U ; ^{238}U is converted into plutonium.

5.2.15 Prior and Present Trends in Nuclear-Fueled Plant Development

The development of nuclear-fueled steam-electric plants underwent substantial change in the 1970s. At the beginning of the decade, orders for nuclear-fueled plants were increasing to a peak of 38 per year. Following the oil crisis of 1973 to 1974, changes in the economy began to affect the cost of, and consequently the demand for, electric power. Opposition to the use of nuclear energy for electric power production increased; litigation was frequently employed. Near the end of the decade, sociopolitical aspects of nuclear-fueled plants became as involved and time-consuming as the technical aspects. In order to participate effectively in the design, construction, and operation of nuclear-fueled plants, one must be familiar with the energy perspective; the concerns about the use of nuclear energy; and the functions of advocates, intervenors, and regulators.

With the maturity of the nuclear-fueled plants, more emphasis was placed on project management (Pederson 1978). Siting of the plants became a major task (Winter and Conner 1978). Because of the reduced demand for electric power, the increased cost of money, and the difficulty of resolving the objections raised, orders for nuclear-fueled plants began to decrease sharply after the middle of the decade. Some orders were canceled. Then in March 1979, a major accident occurred at the Three Mile Island plant, causing serious damage to the plant. This event raised questions about the operation of nuclear-fueled plants and a review of the value of nuclear energy (Rubenstein 1979). At the end of the decade, orders for new nuclear-fueled plants had been reduced to zero and a sizable number of plant orders had been canceled.

The beginning of the decade of the 1980s saw reinforcement of the need for commercial use of nuclear energy (Greenhalgh 1980), but also heralded changes in the safety, control, and maintenance systems. In the electrical area, the most notable changes were the redesign of control rooms and stations and the increased use of computers in more sophisticated safety systems (Hanes et al. 1982). The study of incidents and malfunctions by means of computers has provided another means to inform and guide operators and to evaluate possible trouble spots (Kaplan 1983). The availability and capability of the microprocessor has provided new ways to improve the safety and performance of plant instrumentation, control, and safety systems.

With fewer new nuclear plants being built worldwide than originally anticipated, much attention has been on methods to achieve "life extension" of present plants (retrofitting to allow operation beyond the traditional 20-year life cited for power plants). At the same time, procedures for decontamination and decommissioning of plants being shut down are being refined. The NRC is simultaneously developing streamlined procedures for licensing new plants, with the anticipation that utilities may turn to nuclear energy in the future in the form of the new passive-safe type reactors. This effort, the deregulation of the utility industry in the United States, plus the possible emphasis on nuclear energy as a way to meet goals for reduction of CO₂ greenhouse gases (Schmidt 1998), could have a profound effect on the evolution of the nuclear industry.

There has been a growing belief in recent years that a "rebirth" of nuclear energy has begun. This has been driven by the rapid increase in oil prices coupled with a desire by countries like the U.S. to achieve energy independence, while future energy needs will be met by a combination of conservation plus use of a wide range of energy sources (solar, wind, bio renewable energy sources). There is a growing opinion that nuclear is needed to fill the future needs for central plant energy production. The new "Generation-IV" reactor concepts being developed by DOE offer many attractive features to fit into this need, including competitive costs, passive safety, improved efficiency, reduced maintenance down time and costs, and they also eliminate greenhouse gas emission. The main obstacle remaining is gaining of public acceptance of radioactive waste disposal techniques.

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5.3 NUCLEAR POWER FOR THE FUTURE

By GEORGE H. MILEY

Development in nuclear power for the future is following three main paths: the extension of present reactor designs to create advanced concepts with passive safety features and improved economics; the development of breeder reactors to extend the supply of fissionable fuel; and the development of a new nuclear energy source, fusion. Each of these paths is considered in turn in the following sections.

5.3.1 Advanced Concepts with Passive Safety Features

Recent progress in advanced nuclear power development reveals a high potential for nuclear reactor systems that are smaller and easier to operate than the present generation. Passive, or intrinsic, characteristics are applied to ensure continued cooling of the fuel and its containment systems even in the advent of a major breakdown of the normal cooling/control functions. This substantially reduces the chance of a severe accident. The reactor design concepts that are emerging are simpler, more rugged, have a longer lifespan, and place less burden on equipment and operating personnel. Modular design concepts and design standardization are envisioned to reduce construction time and engineering costs.

Common Development Goals. The primary thrust in U.S. advanced reactor development is to incorporate design improvements to achieve five primary objectives:

Assured safety with features to minimize negative consequences of human error, especially a reduction in the probability of severe core damage.

A significantly simpler design, with increased safety and performance margin in key operational parameters.

High reliability throughout a lifetime of about 60 years.

Reduction in costs to meet the economic competition.

A standardized design which is predictably licensable.

Common generic technical features (passive stability, simplification, ruggedness, ease of operation, and modularity) are being developed to respond to these goals for each of the principal nuclear power systems—the light-water reactor (LWR), the liquid-metal reactor (LMR), and the high-temperature gas-cooled reactor (HTGR). These features, coupled with standardization and assurance that the plant is licensable, should ensure future economic competitiveness.

Light-Water Reactor with Passive Safety Factors. An effort to develop a passively stable LWR in the United States is jointly sponsored by the Electric Power Research Institute (EPRI) and the Department of Energy (DOE) with substantial contributions from several major U.S. suppliers.

Conceptual designs have been developed for passive versions of both a boiling water reactor (BWR) and a pressurized water reactor (PWR). A 600-MWe unit output is proposed to provide the utilities the option of a smaller nuclear power plant and to make it easier and less costly to incorporate passive cooling features.

PWR with Passive Safety Features. An advanced PWR, called AP-600 uses proven technology: a UO_2 -fueled core and field-proven plant components. The burden on systems has been reduced by increasing design margins through reductions in coolant temperature, flow rate, and core power density and by selecting higher-quality materials and more robust components.

Passive cooling is provided by a passive emergency core cooling system (ECCS) and a passive containment cooling system. The ECCS consists of a combination of cooling water sources: gravity drain of water and water ejected from two accumulator tanks under nitrogen pressure. Additionally, the system can be depressurized to permit an even larger amount of water to enter the system. If a feedwater accident disables the steam generators, core decay heat is removed through a passive residual heat exchanger. This transfers core decay heat to the refueling water by natural circulation.

Containment integrity is ensured by cooling the containment shell by evaporating water that is gravity-fed from a large storage tank; heat is removed by a natural-circulation air system. Only automatic valve operations are required to provide emergency core cooling and containment cooling after a major energy release.

It is estimated that the use of a modular design for the AP-600 will reduce construction time to 3 or 4 years. The simplified design combined with the shorter construction time should counter the loss of economy of scale that previously favored larger plants.

The Passively Stable BWR. A passively stable BWR design uses fully proven components and systems operating at reduced burdens: lower power density and increased thermal margin. The reactor operates at full power under natural circulation, which eliminates the need for recirculation pumps while reducing the amount of piping, valves, and controls used in present BWRs.

A passive containment cooling system (PCCS), which consists of three condenser units, provides the capability for removal of the decay heat for a period of 3 days. In the event of a LOCA, the steam discharged from the break and noncondensibles from the dry well flow through the condensers to the wet well. The steam is condensed and the condensate is returned to the gravity-driven cooling system pools. Isolation condensers are used to remove decay heat during reactor isolation events. The condensed steam from the reactor is returned to the reactor vessel, maintaining vessel inventory and limiting the pressure increase without the need to open safety/relief valves. Both the isolation condensers and the passive cooling condensers are located in a large interconnected pool of water on top of the containment building.

The Advanced Liquid-Metal Reactor (LMR). The LMR development work is focused on a modular, passively stable reactor concept called S-PRISM (stable power reactor inherently safe module).

The principal parameters of PRISM are thermal power, 4000 MWt, from four nuclear modules; electric output, 1500 MWe, from two turbine generators; net efficiency, 37.5%; steam conditions, 864°F/2400 lb/in² (absolute); exit sodium temperature, 950°F; core power density, 199 W/cm³; and equilibrium fuel burnup, 150 MW · day/kg. The two reactor modules in each power block provide heated sodium to a single sodium-to-water heat exchanger that generates steam for a single 750-MWe turbine generator. A commercial S-PRISM plant is envisioned to consist of a series of three such 465-MWe power packs, each of which would be functionally independent of the other two.

The reactor vessel auxiliary cooling system (RVACS) provides for emergency core cooling after any incident that causes a loss of the normal emergency heat-conversion systems. The residual-heat removal path consists of radiant-heat transfer from the reactor vessel to the containment vessel where the heat is removed by natural circulation of air between the containment vessel and the biological shield. This combination of passive reactor stability and passive cooling provides assurance of residual-heat removal without operator action.

The reference fuel for the S-PRISM concept is a uranium-plutonium-zirconium alloy with plutonium concentrations of about 25%. An important feature potentially achievable with this fuel is use of a metallurgical processing method for the separation of uranium, plutonium, and the transuranic elements from the fission products. Thus, long-lived transuranics can be recycled in the LMR rather than sent to a disposal site. This approach is basic to the integral fast reactor (IFR) concept pioneered

by the Argonne National Laboratory (ANL). The S-PRISM concept is also capable of using standard oxide fuel in the event that the development promise of the metal fuel is not realized.

ANL has applied its experience in metal fuels operation, fabrication, and reprocessing to develop the IFR concept, which envisions a collocated nuclear power plant, fuel fabrication, and fuel reprocessing center where S-PRISM could function as the power plant. Although such a collocated concept is not essential to S-PRISM, or vice versa, this would provide for greater proliferation resistance because plutonium-bearing materials would not have to be transported outside the security boundaries of the site. The burning of the long-lived actinides via the collocated plant would also significantly ease the waste storage problem.

Advanced Modular Gas-Cooled Reactor. The modular high-temperature gas-cooled reactor (MHTGR) has been the primary focus in advanced gas-cooled reactor development. Its principal parameters are thermal power, 1400 MWt, from four nuclear modules; electric output, 538 MWe from two turbine generators; net efficiency, 38.4%; steam conditions, 1005°F [2515 lb/in² (abs)]; core exit coolant temperature, 1268°F; core power density, 5.9 W/cm³; and equilibrium fuel burnup, 92,200 MW · day/ton.

The reactor core is composed of hexagonal blocks of graphite-fuel elements in an annular array. The fuel is in the form of coated particles of low-enriched uranium oxycarbide and thorium oxide. This fuel and the graphite moderator arrangement have been termed a “prismatic fuel.” It provides an essential barrier to fission product release during an accident. Test data have shown that essentially no failure of the refractory coatings occurs if the fuel is maintained below 1800°C. Even if all active cooling systems were unavailable, decay heat is dissipated by conduction and radiation to the reactor cavity cooling system in the reactor enclosure. This limits the maximum fuel temperatures in accidents to about 600°C, well below the fuel failure temperature.

Tests at a German gas-cooled reactor demonstrated this passive cooling capability after a loss of all coolant flow with no intervention by the operator. The conceptual design of the MHTGR is presently under review by the U.S. Nuclear Regulatory Commission to assess licensability.

Other Advanced Reactor Design Projects. The major industrial countries outside the United States also have advanced nuclear power development programs, several of them larger than those in the United States. The largest programs are in France and the United Kingdom, where the LWR and LMR are under development; and in Germany and Japan, where all three types of reactors are studied. ASEA Brown Boveri in Sweden has proposed a passive LWR, called the process inherent ultimate safety (PIUS) reactor, a concept originating from studies of reactor systems suitable for central heating applications. PIUS is a 640-MWe PWR plant; its core is enclosed in a large prestressed concrete vessel. A fluidic valve is located at the bottom of the core. It introduces, through intrinsic thermal-hydraulic properties, emergency core cooling from the pool of water surrounding the reactor. The concept reduces active equipment beyond the level of U.S. passive designs discussed here.

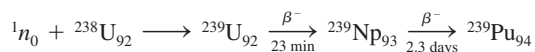
The Canadian nuclear industry is continuing its development of the 600-MWe heavy-water-cooled reactor called CANDU. This reactor has achieved a superior performance record thus far. A 300-MWe CANDU is being developed to provide a smaller-size system for the utilities.

Conclusion. The advanced reactor approach to next-generation reactors will require a substantial development program to verify the safety and cost objectives of the advanced designs. This includes detailed design, licensing review, cost estimates, extensive testing of the LWR and SBWR passive-cooling features, and prototype operations of the modular LMR and HTGR advanced systems. While a variety of concepts have been proposed, they all involve systems that increase the use of passive systems (versus active equipment) to protect against accidents; that is, systems that reduce the dependence on rapid operator response to abnormal conditions, systems that are simple and rugged, and systems that perform reliably without requiring high performance of equipment.

5.3.2 Breeder Reactors

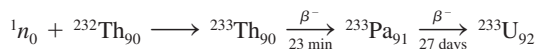
The breeder reactor is able to create more fissionable material than it consumes. A fast breeder is designed to have more nuclear reactions created by neutrons in the high-energy region than by neutrons

in the thermal-energy region, as used by the light-water reactors. This design is accomplished by not moderating, or slowing down, the energy of the neutrons created by one fission reaction prior to subsequent captures of, or fissions by, the neutrons created. In the light-water reactors previously described, the water coolant absorbs energy from the neutrons, creating a condition (called the thermal-energy region) in which more neutrons fission the $^{235}\text{U}_{92}$ isotopes than are captured by the $^{238}\text{U}_{92}$ isotopes. The capture of a neutron by the $^{238}\text{U}_{92}$ isotopes results in a decay chain that produces an atom of plutonium (see reaction chain following). The plutonium can be used to fuel nuclear reactors. Liquid-metal fast-breeder reactors (LMFBRs) operate in the “fast,” or high-energy region, where more of the neutrons are captured by fertile material than cause fission in fissionable material, and where the average number of neutrons created in fission reactions by fast neutrons is greater than the average number created by thermal neutrons. These additional neutrons permit the breeder to produce more fuel by the following reaction chain than is consumed by the fission reactions.



This same reaction occurs in a light-water reactor, but, without the greater number of neutrons and higher ratio of capture cross section to fission cross section, less fuel is created than is consumed. Much of this bred fissionable material is fissioned in situ. For the excess fissionable material in a breeder to be of use in other reactors, or for refueling itself, the spent fuel and blanket assemblies require reprocessing to separate the fissionable material from the fission products created by the nuclear fissions. These fission products, and the damage to the material of the fuel tubes, also limit the life, or time, breeder fuel can remain in the core to produce power. The fission products compete for capture of the neutrons and eventually “poison” the fission reactions, shutting down the chain reaction.

Thermal breeder reactors are possible using a $^{233}\text{U}_{92}$ fuel. This isotope results from the decay chain started from the fertile material thorium, $^{232}\text{Th}_{90}$.



Experimental thermal breeder reactors have been operated using a seed-and-blanket core with water coolant or a molten-salt homogeneous core. The helium-cooled graphite-moderated reactor can also theoretically perform as a breeder.

The present breeder reactor designs use a liquid metal (sodium) to cool the core without moderating the energy level of the neutrons. This liquid metal becomes radioactive (15.5 h half-life) but is isolated from the condensate-feedwater-steam by a secondary, or intermediate, liquid-metal coolant circuit. Thus, an additional heat exchanger and pump are necessary with the attendant impact on plant cost and operating complexity.

Because all fuel resources are finite, breeding of fuels will eventually be necessary to sustain the present use of power with an increasing population and diminishing reserve of fuel. Crude oil and natural gas fuels will be depleted for practical purposes in 100 years. Coal resources will be depleted in about 450 years. Other concerns, such as the greenhouse effects, may limit the use of coal. Without breeding, uranium will be depleted in less than 100 years, as only 0.7% of natural uranium is the fissionable isotope $^{235}\text{U}_{92}$. Breeders will permit most of the uranium and thorium resources to be used as fuel. Breeders can supply our energy needs for thousands of years while avoiding production of greenhouse-effect gases. At a growth rate of 3% per year in power utilization, breeder technology needs to double the fuel every 23 years. Present experimental LMFBRs have demonstrated that *even shorter* doubling times are attainable.

Present Status. In 1983, funds for completing the Clinch River breeder reactor, a power-demonstration fast-breeder reactor, were canceled. Subsequently, the U.S. breeder program has regrouped to focus on the advanced LMR development program described earlier. However, this program was also terminated in the mid-1990s. Russia, France, Germany, Great Britain, and Japan also continued their research and development support of breeder reactors until recent events caused curtailment (but not cancellation) of their efforts.

Breeder Reactor Developments in Europe. Breeder reactor technology is being developed in Europe by combining national with cooperative ventures. Two prototype pool-type plants of 250-MW capacity each, Phenix in France and PFR in the United Kingdom, have been operated. A loop-type plant, SNR 300, was built in Germany but was canceled and terminated. A commercial-size plant, Superphenix, has been in operation in France but is in danger of being decommissioned.

These projects, done on a national basis in the early 1980s, resulted in designs known as Superphenix 2, CDFR, and SNR 2 in France, the United Kingdom, and Germany, respectively. Since 1984, these countries have jointly focused on one single design, the European fast reactor (EFR), which should incorporate the best features from the various national designs.

Breeder Reactor Development in Japan. The LMFBR is being developed in Japan as a national development project. The goal is to develop large oxide-fueled LMFBRs. Japan plans to bring these plants to commercialization by 2020 to 2030.

Joyo, a 100-MWt experimental fast reactor, has been operating successfully since 1977. Monju, a 280-MWe prototype FBR plant, began construction in October 1985, and achieved initial criticality in 1993. The plant was shut down, however, following a 1996 accident.

A demonstration FBR plant, now planned to follow operational experience with Monju, is expected to demonstrate the prospect for commercialization of the LMFBR. A preliminary design study of the demo FBR is in progress in order to verify the technical feasibility of a top-entry loop-type FBR. The plant assumed for this study is to produce between 600 and 800 MWe.

In summary, the Japanese FBR development program is quite aggressive. The Joyo and Monju reactors provide valuable operational experience. A variety of advanced concepts are under active consideration for the next-step demonstration reactor scheduled for operation in the early 2000 time frame so that competitive commercial plants can be introduced by 2020 to 2030.

Breeder Reactor Development in Other Countries. In addition to the U.S., European, and Japanese programs, active work on liquid-metal fast breeders is in progress in the former Soviet Union and India.

Fast-reactor development began in the former Soviet Union in the 1950s. Two reactors, BN-350 and BN-600, are in operation. Design and development of future fast reactor concepts have been initiated with improved safety and better economics as prime goals.

Because natural uranium availability in India is limited, a FBR system has been chosen for increasing the supply of fissile material to meet future energy needs. The fast-breeder test reactor (FBTR) was the first major step in this direction and is to be followed by the construction of a 500-MWe liquid-sodium-cooled fast-breeder reactor, called *prototype fast-breeder reactor* (PFBR). Design and construction of PFBR is expected to be the beginning of the commercial deployment of such reactors.

The FBTR attained its first criticality in October 1985 and began producing electricity in 1991. The first core is small, producing only 16 MWt. The next core to be introduced will be close to the reference design of about 35 MWt. FBTR has served the purpose of a pilot plant and has also turned out to be a good investment for developing trained workers. This experience will lead to the next significant step: the 500 MWe PFBR.

Projections for future energy requirements have driven the India breeder program. It is estimated that India will need over 3000 billion units of electricity by 2040. The goal for FBRs is defined by the need to supply at least 1000 to 2000 billion units so as to significantly reduce the use of coal for electrical production. An installed capacity of 200 to 400 million kW in fast breeder reactors would be required to meet this goal. Starting with an annual feed of 3000 kg of plutonium produced from 10 million kW of PHWR capacity, the realizable growth for different fuel options is given in Table 5-5.

Thorium is not introduced in the fuel cycle in any significant way. Thus, the fast reactor program would be based on Pu-U-238 cycle as long as depleted uranium is available as a by-product of the first-stage heavy-water program. Preference for depleted uranium is based on higher growth potential of a Pu-U-238 fuel cycle.

Table 5-5 shows that a ternary metal alloy comes closest to meeting India's needs when the fuel is designed to give a specific power of about 0.3 MWe/kg of fissile material and a breeding ratio of 1.4 to 1.5, corresponding to an annual net breeding gain of 316 kg/GWe. Advances reported for the

TABLE 5-5 Growth of Fast Breeder Installed Capacity

Characteristic	Fuel type (U + Pu)		
	Mixed oxide	Mixed carbide/nitride	Metal
Specific power MWe/kg	0.25	0.30	0.30
Breeding gain, kg/GWe · year, allowing for processing losses	130	188	316
Installed capacity, GWe · year			
2010	7.8	10.4	11.6
2030	30.8	54.6	90.6
2050	53.2	150.2	430.4

Note: 3000 kg/year fissile material from 10000-MWe PHWRs; 3 years for recovery of bred material; process loss per GWe/year, 14 kg oxide and 28 kg carbide and metal.

development of the ternary metal alloy make the assumed figures appear realistic. Research and development for fuel development has therefore shifted from the mixed carbide to the ternary metal alloy.

The Indira Gandhi Center for Atomic Research and Nuclear Power Corporation of India Ltd. have created a new public sector company Bharatiya Nabhikiya Vidyut Nigam Ltd. to carry out the construction and operation of the India Department of Atomic Energy's first 500-MWe prototype fast breeder reactor.

5.4 NUCLEAR FUSION

By GEORGE H. MILEY and STEPHEN O. DEAN

The substantial, research effort now in progress aimed at controlling thermonuclear reactions to produce electric power at gigawatt levels is stimulated by the fact that the basic fuel consumed in fusion, deuterium, naturally occurs to the extent of one atom for every 6500 atoms of hydrogen, making ocean and lake water the basic fuel. This represents a virtually inexhaustible fuel source. Fusion liberates energy by combining two nuclei of light elements into one nucleus of a heavier element. The resulting mass is less than that of the fusing nuclei. Fission, results in splitting one atom of a heavy element into two atoms of lighter elements. The resulting mass, again, is less than that of the fissioning atoms. Each fusion reaction releases about 20 MeV of energy. Each fissioning reaction releases about 200 MeV of energy. (See also Sec. 5.4.)

Fusion will enable use of an extremely large fuel resource—the deuterium isotope, $^2\text{D}_1$, which exists as 1 part in 6500 parts of $^1\text{H}_1$ in natural hydrogen. The deuterium available from the ocean could fuel the power requirements of the world for billions of years—even until the Sun expands and envelops the Earth. However, research is still far from developing a commercial fusion reactor. Fission, solar energy and coal, other renewables, and nuclear fusion provide the means of avoiding severe shortages of electricity and other forms of power in the twenty-first century.

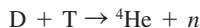
Many years will be required to develop a commercial fusion power plant. Current estimates for the introduction of commercial fusion plants are generally for sometime in the mid-twenty-first century. In fiscal year 2004, the Dept. of Energy outlay in support of civilian fusion research in the United States exceeded \$263 million. A larger budget, \$514 million, is provided by the National Nuclear Security Administration for inertial confinement fusion aimed at simulating nuclear weapons.

5.4.1 Fusion Reactions

The reactions of interest for fusion power occur when nuclei (ions) of elements of low atomic number are brought together in a plasma, at such temperature (i.e., velocity) as to give the nuclei sufficient speed to overcome the repulsive coulombic barrier between them (and fuse), thereby transforming a

part of their mass into kinetic energy. Because it is most reactive, the reaction generally favored for first-generation fusion plants involves deuterium, tritium, and lithium, in the so-called D-T-Li cycle.

This cycle involves two types of reactions. In the first, deuterium and tritium react to produce an alpha particle (helium-4 nucleus) and a high-energy neutron:



The helium-4 ion emerges with an energy of 3.5 MeV, and the neutron with 14.1 MeV.

The second reaction involves the production of tritium by lithium, which is contained in a “blanket” surrounding the plasma. Two neutron-induced reactions are involved in the production of tritium:



and



The tritium used in the D + T reaction is thereby regenerated by the lithium reactions in the “blanket” region external to the plasma. The power density produced by this reaction reaches a maximum when the kinetic temperature of the plasma is about 15 keV.

5.4.2 Advanced Fuels

A number of fusion reactions are possible at temperatures of potential interest for fusion (<1 MeV) that involve various light elements through boron. The principal advantages of seeking such “advanced fuels” include elimination of the need to breed tritium, reduced neutron fluxes (to reduce both materials damage and induced radioactivity in vessel and blanket structure), and an increased fraction of fusion energy carried by charged particles (making direct energy conversion attractive). Two general classes of fuels are of interest: deuterium-based and proton-based. Examples of the former include deuterium-deuterium (D-D) and deuterium-helium 3 (D- ${}^3\text{He}$) fusion while hydrogen- (proton)-boron 11 (p - ${}^{11}\text{B}$) is a frequently cited proton-based reaction. The deuterium-based reactions all involve neutron production to some degree due to D-D reactions whereas p - ${}^{11}\text{B}$ represents a fairly ideal fuel since the primary product is helium ($p + {}^{11}\text{B} \rightarrow 3\alpha$). The supply of hydrogen and boron 11 (from borax) is virtually as extensive as for deuterium; ${}^3\text{He}$, however, must be bred, either via D-D reactions or through the decay of tritium (12-year half-life). Another potential source of plentiful ${}^3\text{He}$ that has gained considerable attention involves lunar mining. Bombardment of the lunar surface by the solar wind has resulted in the impregnation of the upper crust with a variety of gases, including ${}^3\text{He}$.

Because of the relatively small fusion cross sections involved and the higher energies (plasma temperatures) required, all of the advanced fuels pose more stringent confinement requirements and offer lower power densities compared to D-T-Li fusion. Consequently, these fuels appear to be candidates for later-generation power plants. Still, the improved environmental compatibility they offer, combined with a significant reduction in neutron damage to structural materials, make this an important long range goal.

5.4.3 Power Production

The major part (about 80%) of the energy released by the D-T-Li cycle is carried by the high-energy neutrons. These, being chargeless, cannot interact with electric or magnetic fields, and hence cannot produce electric power directly. Rather, their power is converted to heat, which is extracted from the lithium blanket by thermal transfer. This heat (like that in fission- and fossil-fueled systems) must be converted to electricity by a thermal system, for example, one employing vapor (e.g., steam) or a working liquid (e.g., potassium).

The remaining 20% of the power generated involves charged alpha particles (helium-4 nuclei). This power remains in the plasma, thus sustaining its temperature. With a p - ${}^{11}\text{B}$ cycle, the charged particle energy could be increased to nearly 100%, making direct conversion a more important avenue for energy extraction.

Energy Breakeven. In nuclear fission, the reaction is neutron-induced and self-sustaining when criticality is reached. In nuclear fusion, on the other hand, the strong mutual repulsion between the positively

charged deuterium and tritium nuclei must be overcome to cause them to fuse. To do so, power must be supplied to start and heat the plasma to create reactions. The power derived from the fusion reactions must, of course, exceed the input power. Energy breakeven is the condition at which the fusion energy released just equals the energy input required to heat the plasma to the fusion temperature.

To achieve energy breakeven, the product $n\tau$ of the plasma density n and the time τ during which the plasma is confined must lie within a narrow range for a specific reaction. In the D + T reaction this product (i.e., $n\tau$) must be roughly 10^{14} ions/cm³ · s (frequently called the “Lawson criterion”). Thus, if confinement is achieved for a few tenths to several seconds (typical of magnetic confinement), the plasma density required lies in the range of 10^{14} to 10^{16} ions/cm³. If the confinement time is of the order of a thousandth of a microsecond (typical of inertial systems), much higher densities, of the order of 10^{26} ions/cm³, must be achieved. In all cases, the plasma temperatures must be in the range of 10 to 15 keV.

Power Balance. To obtain useful amounts of power from fusion, not only must the output fusion power meet the conditions for power breakeven and all the other power inputs required by the reactor, but the fusion-released power must exceed this input power by a large amount, the difference being equal to the useful output. The power input to the auxiliaries of the fusion reactor, which heat and confine the plasma, is typically 10%–20% of the total power.

Since useful power output from a fusion reactor can be obtained only when substantial input power is provided, the reactor must be viewed as a power-amplifying device, the degree of power amplification differing according to the plasma-confinement scheme and the design of the lithium blanket, but typically is on the order of a few hundred.

5.4.4 Nonelectrical Applications

In addition to electrical production, fusion reactors can potentially be used for a variety of other important purposes. For example, neutrons from D-T (and other D-based fuels) can be employed very effectively in a fusion-fission hybrid to breed fissile fuel in the blanket region of the fusion device. A unique feature of these hybrids is that very high “support ratios” are conceivable; for example, with some concepts, as much as 20 times the energy production of the fusion plant can be provided in light-water fission plants. Fusion neutrons may also be used in conjunction with fission processes in hybrid systems and to accelerate the decay of fission wastes, that is, providing a fission waste burner. Several methods for chemical production by fusion plants have also been studied. These generally use the penetrating power of the high-energy neutron to create a high-temperature region in the blanket. This then appears well suited to various processes, for example, hydrogen production from water via either high-temperature electrolysis or thermochemical processes. The use of radiation, neutrons, or the plasma itself has also been considered for chemical processing, but this approach is not as straightforward (or as developed) as thermal processing.

Fusion driven propulsion units for deep space missions have also received serious attention. In this case, the fusion heated plasma can be exhausted through a magnetic nozzle to achieve directed thrust. This gives very high exhaust velocities (or high “specific impulses”) making fusion propulsion extremely well suited for space missions to Mars and beyond where fast transit times are essential.

5.4.5 Plasma Confinement

The high temperatures and pressures of the fusion reaction preclude the use of a material vessel to confine the plasma during the reaction. Two primary approaches are now under investigation: magnetic and inertial confinement.

In the magnetic confinement scheme, intense magnetic fields are generated in the reactor, oriented with respect to the reaction space such that the plasma ions and electrons experience an inward pressure that resists the outward pressure of the hot plasma.

In inertial confinement, D-T-fueled targets (e.g., small plastic ampules) are bombarded by intense pulsed laser, x-rays, or charged-particle beams. Many beams are arranged to impinge symmetrically on the target from different directions; that is, beams are aimed at a common point at which the target is positioned. Each beam is pulsed and rapidly heats the surface of the target. A very high energy, exceeding a megajoule, is required in each pulse. Typically, each pulse lasts about 10^{-9} s and the pulse rate is about 1/s. The symmetrical heating at the surface of the pellet causes it to ablate. This creates

a strong inward force such that the target's interior is compressed. Fusion then occurs at the center and the reaction burns outward. High-energy, heavy ions are considered to be potentially attractive for inertial confinement fusion, the key advantages being the high efficiency of these ion accelerators and improved coupling of the beam energy into the target compared to photons (lasers). Also, the potential for the scale-up of present accelerators to achieve economical, efficient, high-repetition-rate units seems good based on developing accelerator technology.

Magnetic Confinement. The principal means for magnetic confinement is the tokamak. Other approaches include, the reversed field pinch, the spheromak, the tandem mirror, and the stellarator. However, the level of support for research on alternative approaches is only a small fraction of that in tokamaks.

A measure for the performance of the magnetic confinement methods can be expressed by a parameter β , the ratio of the outward plasma kinetic pressure to the inward confining pressure of the magnetic field. This parameter denotes the efficiency of the magnetic confinement; that is, high- β systems make better use of the confining field than do the low- β systems. The parameter β is defined as

$$\beta = \text{constant} \times \frac{n(T_i + T_e)}{B^2}$$

where n is the plasma density, T_i and T_e the ion and electron kinetic temperatures, and B the confining magnetic field strength. High- β reactors are expected to operate with β greater than 0.8, and low- β designs at values below 0.2. The fusion power density varies as the square of β . Tokamaks are generally low- β devices. Spheromaks and reversed-field configurations are examples of high- β devices.

Inertial Confinement. Originally termed "laser fusion" this confinement method focuses a tremendous pulse of energy onto the surface of a sub-millimeter scale target containing fusible fuel, e.g. D and T. This energy causes the outer layer of the target to be ablated away in "rocket exhaust" fashion. The resulting inward directed force causes compression of the target to ultra-high densities, approaching a thousand times solid density. In the process, compression heating raises the target fuel to fusion reaction conditions. A point is reached where the internal pressure of the hot, high density target just balances the internal compressive force. This results in a "stagnation" point in the inward trajectory of the target radius, followed by a rapid outward expansion. The time associated with this motion around a stagnation point can be viewed as the confinement time, τ . In effect the target dynamics is governed by the inertia of the individual target ions, hence the terminology "inertial confinement fusion". A rough estimate of τ is given by R/v_{th} where R is the compressed target radius and v_{th} is the average speed of the target ions. Then if $R \sim 10^{-6}$ m and $v_{\text{th}} \sim 10^3$ m/sec, $\tau \sim 10^{-9}$ sec. While very short, this confinement time proves to be adequate for net energy production.

The key point is that with the very high density, the fusion reaction rate is so large that this time is sufficient to produce more fusion energy than that required for the compression. In addition to lasers, it is now recognized that other pulsed energy sources (termed "drivers") such as x-ray producing Z-pinches and heavy ion accelerators can focus the required energy on these extremely small inertial confinement targets.

Other Confinement Methods. While magnetic and inertial confinement are the most widely considered approaches, a variety of other techniques have (or are being) considered. Examples include high speed acceleration of a target against a "wall" or another projectile, creating "impact" fusion. Electrostatic fields can not confine a plasma in steady state, but if directed ion motion is created, the ion inertia itself prevents directed plasma losses, providing Inertial Electrostatic Confinement (IEC). The replacement of an electron in liquid deuterium or tritium with a muon (i.e. a "heavy electron") results in a much smaller orbital radius for the muonic deuterium atom. This in turn allows atoms to approach each other at shorter distance than normal and thus permits fusion in the liquid (or in effect "cold fusion") without requiring the high ion velocities of a hot fusion. This is termed muon catalyzed fusion. These examples are far from being inclusive but they indicate the "richness" of the field.

5.4.6 Tokamaks

In tokamaks, the plasma-confinement space has closed, toroidal geometry, as shown in Fig. 5-24. A current in the toroidal field coils produces a toroidal field of the order of 100 kG, which confines the

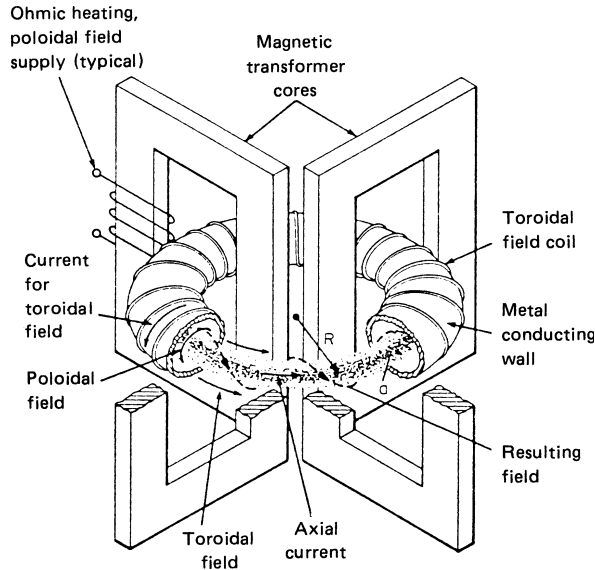


FIGURE 5-24 Elements of the tokamak reactor. Two magnetic fields (toroidal and poloidal) combine to produce a resulting field shown in the dashed line. This field both confines the plasma and causes its ions and electrons to move through the cavity. The ohmic heating of this axial current is one component of the heat required to achieve fusion of the plasma. Additional heating must also be supplied from an external source to maintain the fusion temperature in excess of 100 million degrees Kelvin.

plasma within the toroidal cavity. A second magnetic field, the poloidal field, is produced by external magnetic cores so that this field falls at right angles to the toroidal field everywhere within the cavity. The vector sum of the two fields (dashed line in Fig. 5-24) causes the plasma ions and electrons not only to be confined but to move through the torus, producing an axial current of tens of hundreds megaamperes. The ohmic heating associated with this current is an important source of power input for heating the plasma. However, in a tokamak, ohmic heating alone is insufficient to produce fusion temperatures. Additional plasma heating must be supplied by one of various methods, or combinations of them, such as radio-frequency (rf) heating, adiabatic compression, or the injection of high-energy beams of fuel particles.

A third, transverse, magnetic field is also needed to provide control of the position of the plasma column within the cavity. Additional magnetic coils may be involved to create a magnetic “divertor” which is used to scrape off the outer surface of the plasma and divert it into a “dump” vacuum pump region. The divertor serves three key functions: removal of impurity ions attempting to enter the plasma from the plasma chamber’s wall, removal of fuel and fusion products that diffuse to the outer surface of the plasma, and removal of heat carried by escaping plasma.

When operating in the burn cycle, the kinetic temperature of the plasma is expected to range from 10 to 30 keV; the latter figure is above 300 million degrees absolute. Several large experimental devices, for example, the JET tokamak in England and the TFTR tokamak at Princeton University, have achieved temperatures in the range of 10 keV, and near energy breakeven conditions. In pulsed operation of the tokamak, successive periods of plasma confinement will last several seconds or tens of seconds. The total burn time, prior to purging and reloading the reactor, will be of the order of several hundreds or thousands of seconds. Auxiliary heating power of 10 to 100 MW is required to produce a fusion power output of 1 to 5 GW. Recent studies have concentrated on the possibility of using rf or other techniques to provide a current drive which can ultimately permit steady-state rather than pulsed operation. Deuterium and tritium fuel would be injected at a rate of about 2 to 4×10^{22} atoms/s to provide a density of about 10^{14} ions/cm³ in a low- β (3% to 10%) tokamak.

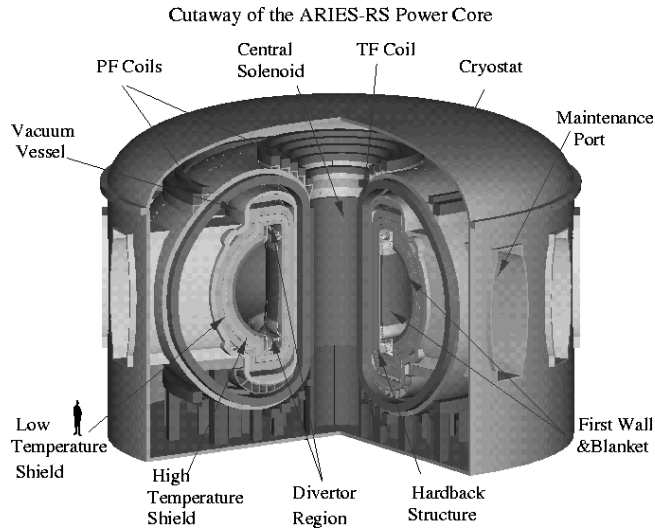


FIGURE 5-25 Schematic diagram of the ARIES-RS fusion power core unit. (Najmabadi and the ARIES Team 1997.)

A number of conceptual studies of the prospective operating characteristics of tokamak reactors have been made. Figure 5-25 and Table 5-6 are the results of such a study done by the ARIES study team at the University of California, San Diego, along with other university and industrial collaborators (Najmabadi 1997).

Figure 5-26 shows a schematic drawing of the TFTR which was designed to study burning D-T plasmas. Built at the Princeton Plasma Physics Laboratory (PPPL) at a cost of over \$500 million, this

TABLE 5-6 Operating Parameters of the ARIES-RS Tokamak Power Plant

Aspect ratio	4.00
Major radius, m	5.52
Minor plasma radius, m	1.38
Plasma vertical elongation (X point)	1.70
Plasma current, MA	11.32
Bootstrap current fraction	0.88
Current-drive power, MW	81
Toroidal field on axis, T	7.98
Peak field at TF coils, T	16
Toroidal beta	0.05
Average neutron wall load, MW/m ²	3.96
Primary coolant and breeder	Natural lithium
Structural materials	Vanadium and steel
Coolant inlet temperature, °C	330
Coolant outlet temperature, °C	610
Fusion power, MW	2170
Total thermal power, MW	2620
Net electric power, MW	1000
Gross thermal conversion efficiency	0.46
Net plant efficiency	0.38
Recirculating power fraction	0.17
Mass power density, kWe/metric ton	66.70
Cost of electricity, mill/kWh	75.79

Source: Najmabadi and the ARIES Team (1997).

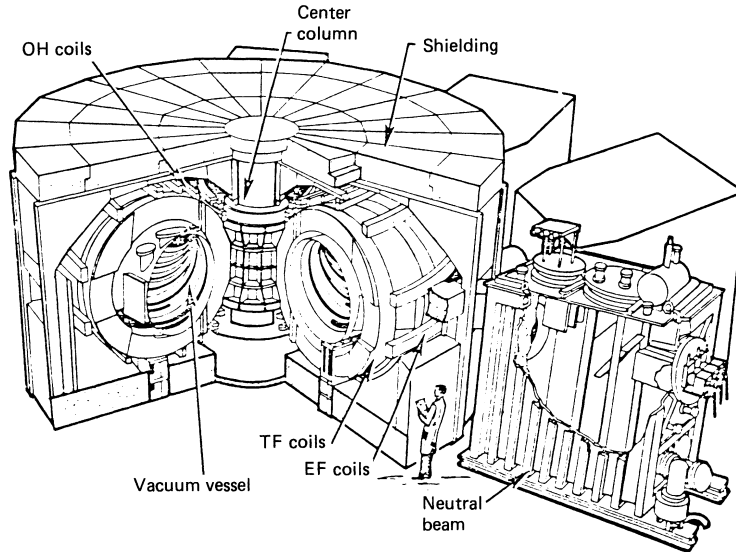


FIGURE 5-26 Schematic drawing of TFTR.

device began initial operation in December 1982. Since then, the TFTR has carried out a series of impressive experiments. Objectives have been to obtain improved confinement, a basic understanding and control of instabilities, improved heating and current drive, impurity control, and plasma-wall interactions. These results have been complemented by a number of other major tokamak experiments worldwide. These included D-III (San Diego, USA), JET (Culham, UK), JT-60 (Japan), and T-12 (Russia). A major step in this work involved the study of burning D-T plasmas in both TFTR and JET. This provided valuable experience with a fusion plasma where a major fraction of its heating comes from energetic fusion products (the 3.5-MeV alpha particle in the case of D-T fusion). It also provided experience with safe handling and control of tritium. Then, after several successful campaigns with D-T burns, Congress cut funding for TFTR stating it had met the design objectives. The experiment was then shut down in 1997.

The TFTR employed ~30 MW of neutral beam heating in a device with 12.48-m major radius (R) and a 0.85-m minor radius (a) and an on-axis toroidal field of 5.2 T. It produced plasmas with densities of 10^{13} particles/cm³, plasma temperatures of 10 keV, and fusion power densities of the order of 1 MW/m³. The JET experiment is still in operation.

5.4.7 World Facilities for Fusion Research and Reactor Concepts

A large number of both magnetic and inertial fusion research facilities are in operation worldwide. A complete list of institutes, ordered by country, city, and acronym, along with a listing of major equipment and scientific staff, is available from the International Atomic Energy Agency, Vienna (World Survey of Activities in Controlled Fusion Research, 2001 ed.).

Magnetic Confinement Facilities and Concepts

Operational Experience with Large Tokamaks. Extensive operational experiences have been obtained with hydrogen, deuterium, and D-T plasmas in major tokamaks around the world. Two—TFTR and JET—have used D-T plasmas. This experience, plus some data about prior operations, is provided here for these two major tokamak facilities.

TFTR operational experience is described in detail by von Halle (1998) and Johnson (1995).

In April 1997, the TFTR completed its final operating period, bringing to a close a highly successful phase of fusion research. The TFTR produced over 80,000 high-power plasmas since 1982 with the

objectives of studying the plasma physics of large tokamaks, gaining experience in the solution of engineering problems associated with large fusion systems, and demonstrating fusion energy production from the burning, on a pulsed basis, of deuterium and tritium in a magnetically confined toroidal plasma system.

In 1993, TFTR became the first magnetic fusion device to study plasmas using nearly equal concentrations of deuterium and tritium. Since that time, over 1000 D-T experimental shots and over 23,000 D-D shots were carried out, demonstrating new regimes of plasma confinement, proof of alpha heating, and reactor-level fusion power densities by producing a plasma that yielded over 10 MW of fusion power at a corresponding central fusion power density of approximately 2.8 MWm^{-3} (Nazikian 1996, Bell 1997).

The TFTR technical systems routinely operated at or beyond the original design criteria throughout the period, maintaining an impressive machine availability of $>85\%$. This success continued through the final night of operations when a plasma with a record 7.7 MJ of stored energy was attained.

An important additional contribution from this work was the demonstration of safe operation of the tritium systems, with over 950 kCi of tritium processed within the constraints of a 50-kCi site limit and a 20-kCi machine limit. The tritium purification system “closed the loop” on the TFTR fuel cycle during the final operating campaign, processing >50 kCi of tritium at $>95\%$ purity back to the tritium storage and delivery system U-beds for further use on TFTR experiments.

JET operational experience is outlined by Bertolini and The JET Team (1997). JET started operation in June 1983, as the largest tokamak experiment of the coordinated fusion research program of the European Union. The global objective of JET was to produce and study plasmas of thermonuclear grade, in configurations suitable for extrapolation to a reactor. This led to the early choice of machine parameters with D-shaped toroidal coils, vacuum vessel, and plasma cross section. Great flexibility and suitable stress margins were included in the original JET design, to allow modifications and/or upgrading of the machine to follow the rapidly evolving requirements of the physics program (Keilhacker and The JET Team 1997).

Two major interventions took place between 1986 and 1989. The first involved increasing the plasma current capability from the design value of 4.8 to 7.0 MA in limiter configuration and from 3.0 to 5.0 MA in the X-point configuration. This work involved revising the design of the toroidal and poloidal coils, of the mechanical structure and of the vacuum vessel, by detailed finite element computer modeling and calculations, and fatigue tests on the prototype toroidal coil. Moreover, new power supplies had to be provided and plasma control also had to be improved, to cope with the enhanced vertical instability of the plasma ring. This upgrading was generated by initial experimental results, which showed a sharp decay of the energy-confinement time with heating. This could be counteracted by increasing the plasma current and by setting up an X-point configuration, which allowed H modes to be established.

The second change required progressively covering the inconel vessel walls with low-Z materials, specifically, graphite tiles ($Z = 6$) and later beryllium tiles ($Z = 4$), supplemented at first by wall carbonization and later by beryllium evaporation. This intervention was prompted by the need to substantially reduce Z_{eff} in order to decrease radiation losses.

At this point, the JET plasma could be maintained for only about 1 s, limited by a combination of MHD instabilities and accumulation of impurities in the X-point region. Active control of the impurities was therefore required, and was achieved by installing a pumped divertor to control impurity level and particle and energy exhaust and to enhance plasma energy confinement in H mode (Bertolini 1995). This represents the third major upgrading of the JET tokamak. The original features of the JET design allowed the basic structure of the machine to be maintained (viz., toroidal and poloidal coils, mechanical structure, and vacuum vessel). However, the necessity to install the divertor coils inside the vacuum vessel produced a loss of about 25% of the plasma volume, but it allowed plasma currents up to 6 MA to be accommodated.

Three divertor configurations (Mark I, Mark II, and Mark III Gas Box) have been designed, with progressively more closed configuration to enhance particle and impurity retention in the divertor chamber and to increase the amount of plasma energy released by radiation. The results of JET divertor studies are of great importance to finalize the ITER divertor design.

The full implementation of these measures led to the development of the hot-ion regime, showing a spectacular increase in overall plasma performance. The fusion triple product improved from

0.12 to $0.9 \times 10^{21} \text{ m}^{-3} (\text{s} \cdot \text{keV})$ and the equivalent energy gain Q_{DT} increased from 0.01 to 1.07. In separate pulses, an ion density of $n_D \sim 4 \times 10^{20} \text{ m}^{-3}$, an ion temperature $T_D \sim 30 \text{ keV}$, and an energy-confinement time of $\tau_E \sim 1.8 \text{ s}$, were obtained.

Finally, a $Z_{\text{eff}} \leq 2$ and a dilution factor $n_D/n_e \geq 1.9$ were reached. This level of global plasma parameters was sufficient to perform the first ever D-T experiment, which, in spite of using a mixture far from optimum (11% T–89% D), achieved a peak fusion power of 1.7 MW with >50% of thermalized neutrons, and a fusion energy of 2 MJ (The JET Team 1992). The additional heating power has been progressively increased up to 50 MW (20 MW of neutral beam injection, 20 MW on ion-cyclotron resonance heating, and 10 MW of lower hybrid). These results of internal plasma-wall components, especially the divertor, are under way with a new D-T experimental campaign planned for fall 1999 (The JET Team 1995).

A recent series of experiments with deuterium-tritium plasmas (DTE1) are of particular importance for fusion reactor developments (Stork and The JET Team 1997). These are the first plasma experiments with a 50:50 D-T mixture in a tokamak with a divertor.

Extensive work was undertaken to ensure that the JET machine and its major subsystems were able to safely carry out an extended period of D-T operation (Hemmerich 1989). This included adding the JET active-gas-handling system (AGHS) and its commissioning to full closed-cycle operation. As of 1998, the AGHS supplied around 40 g of tritium to the JET torus and neutral-beam injectors (NBIs) and reprocessed batches of tritium of over 11 g returned from the cryopumps of the torus and NBI.

Several modifications were required to bring the NBI system to full tritium compatibility (Stork and the JET Team 1997). The injection of tritium beams at energy up to 155 kV and total power of up to 11.3 MW has taken place.

Prior to D-T operation, JET was subjected to extensive deterministic analysis of design-basis accidents (DBAs) to establish changes required to protection systems and to satisfy the regulatory authorities.

A planned intervention to repair a small water leak in the tritium neutral injector was successfully undertaken during these operations (Stork and the Jet Team 1997). The exhaust detritiation system (EDS) was effective in keeping environmental discharges to below management limits throughout operations.

Other Major Tokamak Experiments. Although not all-inclusive, this section indicates the status and diversity of tokamak research worldwide.

The JT-60 tokamak, located at the Naka Fusion Research Establishment, Ibaraki-ken, Japan, is one of the world's largest tokamak experiments (Neyatani and The JT-60 Team 1995, Sakasai and The JT-60 Team 1997). To extend the earlier results from JT-60U, a superupgraded design (JT-60SU) has been developed. It has a superconducting toroidal field and poloidal field coil system with a pulse length of 2000 s or more. Figure 5-27 compares cross-sectional views of JT-60SU and JT-60U.

Typical operation parameters are shown in Table 5-7. The physics goal is a simultaneous achievement of stable steady-state full current drive plasma with high confinement and high bootstrap current fraction, together with a dense, cold, radiation-divertor function in reactor-relevant conditions.

For steady-state physics, the pulse length is set to exceed 2000 s, which is sufficiently longer than the characteristic times of the particle saturation of the wall (30 to 60 s), current diffusion (100 to 400 s), and first-wall temperature (500 to 1000 s). Establishment of integrated physics, technology, and engineering for long-pulse operation under steady-state reactor-relevant conditions is also important for future steady-state reactors, such as developing superconducting coils, and high-heat-conductive secondary gamma-ray shielding. Progress in high-performance and steady-state experiments on JT-60U during 1995 to 1997 is summarized as follows:

JT-60U research on the steady state with high fusion performance focused on contributions to ITER physics R&D. Experiments on negative-ion-based neutral beam injection (N-NBI) indicated a high neutralization efficiency of 60% at 370 keV and a high current drive efficiency of $\eta_{CD} = 8 \times 10^{18} \text{ m}^{-2} \cdot \text{A/W}$.

Toroidicity-induced Alfvén eigenmodes (TAE modes) excited by N-NBI were observed for the first time. The divertor was modified from the open divertor to the W-shaped pumped divertor in February to May 1997.

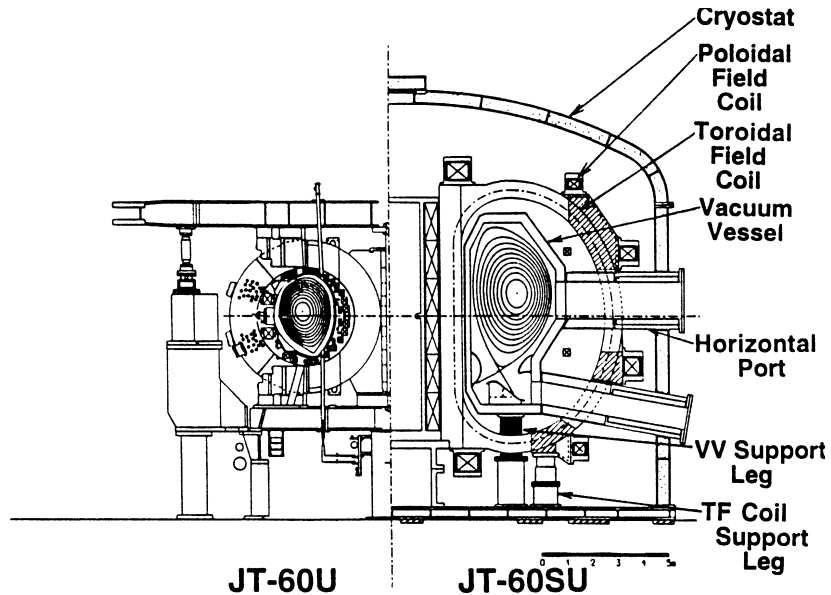


FIGURE 5-27 Cross-sectional views of JT-60SU and JT-60U.

In the new pumped divertor, radiative divertor by neon and deuterium injection with high recycling enhanced the divertor radiation with helium pumping in ELMy H-mode discharges.

Halo current was measured in the new divertor, and a new database was added for ITER. A high equivalent Q_{DT} of 1.05 was achieved in high-performance reversed magnetic-shear discharge.

Quasi-steady reversed-shear plasma with internal transport barrier and H-mode edge was sustained for 4.3 s. High-triangularity shaping was effective for improvement of the giant ELM limit and β_{N^*} . Steady-state high performance in the ELMy H mode was sustained for 9 s with the new divertor.

TABLE 5-7 Main Parameters of JT-60SU

Toroidal field strength B_1 at 4.8 m, T	6.3
Plasma current I_p , MA	10
Major radius R_p , m	4.8
Minor radius a_p , m	1.3–1.5
Aspect ratio	3.2–4
Elongation κ	1.8
Triangularity δ	0.4–0.8
Number of TF coils	18
Total volt-seconds, V · s	170
Pulse length, s	>2000
NB power P_{NB} , MW	65
LHRF power, MW	20
ECRF for breakdown, MW	1

The DIII-D Tokamak. The mission of the DIII-D national program, centered at General Atomics in San Diego, Calif., is to establish the scientific basis for the optimization of the tokamak (Calis et al. 1989, Petersen and The DIII-D Team 1997). The research is carried out in four areas—transport, stability, boundary, and current drive and heating—in collaboration with a large number of national and international collaborators. The main goal is to optimize the performance of the tokamak through active control of the plasma shape and the plasma profiles. In the transport area, the emphasis has been on the study of the role of internal transport barriers in improving confinement. In the stability area, the emphasis has been on the study of neoclassical MHD, resistive wall stabilization, density limits, and disruption characterization and mitigation. In the boundary area, the emphasis has been on developing understanding of radiative divertor physics and the initiation of the new radiative

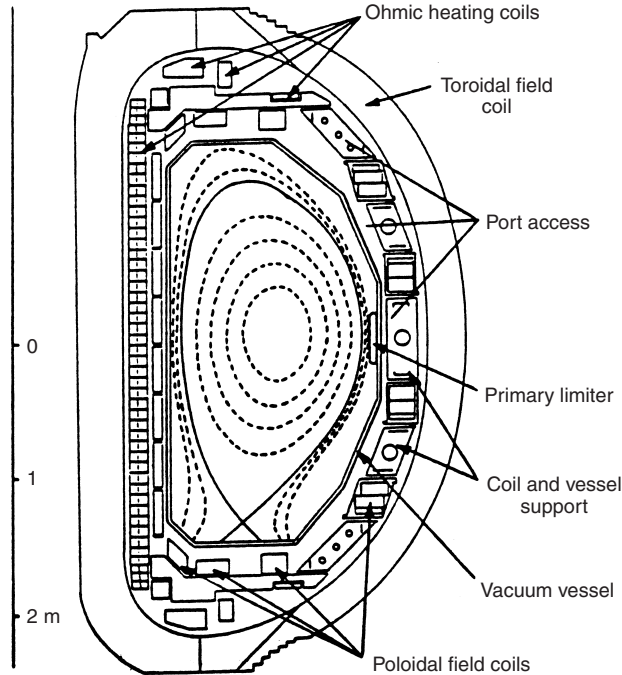


FIGURE 5-28 Cross section of the DIII-D device. Superimposed is the MHD equilibrium of a high-beta divertor.

divertor, and in the current drive and heating area on the preparation of two 110 GHz gyrotrons. These areas will be discussed in more detail in the following sections.

This research program utilizes the DIII-D tokamak (Fig. 5-28), which is a mid-sized device operating at near-reactor-level plasma parameters. It has a D-shaped cross section with an aspect ratio of 2.5:1 and a major radius of 1.6 m, and is capable of producing a large variety of plasma shapes, including elongation up to 2.6 and triangularity to 1.0. The maximum toroidal field on axis is 2.1 T, and the maximum plasma current is 3 MA. The shaping flexibility is due to the 18 fielding shaping coils that are located close to the plasma and individually controlled. A divertor baffle and a cryopump are installed above the floor and at the ceiling of the DIII-D vessel to allow study of divertor physics and control the plasma density. The auxiliary heating of the plasma is provided by eight neutral beams producing 20 MW at 80 kV in deuterium, 1.5 MW (source) of 110 GHz electron cyclotron heating, and 6 MW (source) of fast-wave current drive.

More than 50 different diagnostics are used to probe the DIII-D plasma. The Thomson scattering system measures the electron temperature and density every 2 ms at 40 different locations including the divertor region. The rotational Stark effect diagnostic measures the local pitch angle of the magnetic field, and thereby of the plasma current density, and the radial electrical field. The charge exchange recombination diagnostic measures ion temperature, poloidal and toroidal rotation, impurity density, and radial electrical field. There are several diagnostics that measure fluctuations (beam emission spectroscopy, far infrared scattering, etc.). A digital control system is used to control the plasma shape, current, and density and the injection of auxiliary heating power.

Increasing electron cyclotron heating (ECH) has high priority. With the successful testing of the two gyrotrons and development of new diamond windows, it appears that long-pulse gyrotrons are now available. The ECH power will be used for current profile control, perturbation studies, and transport barrier control. An extension of the DIII-D pulse length can be obtained by installing a new return bus for the toroidal coil, with minor upgrades of the field-shaping coil supplies. A longer tokamak

pulse duration is important for stability studies, extension of the advanced tokamak mode, and wall stabilization. Upgrade of the Thomson scattering system to include a central cord is important for transport and neutral density measurements, understanding of disruptions, and to aid active mode control of neoclassical MHD modes. The top divertor will be completed with inner and private flux baffles, which should help reduce core fueling of neutrals and thereby improve confinement and increase the electron temperature for a more efficient current drive. Later, the lower divertor will be upgraded for advanced tokamak operations.

Tore Supra. Among the large tokamaks in operation around the world, Tore Supra (Centre d'études de Cadarache, France) has the unique feature of a superconducting magnet which provides a permanent toroidal field. With its niobium-titanium coils operating at 1.8 K, the Tore Supra toroidal field is routinely set up for 4-T/(12-h/day) operation. Its main research directions are therefore concentrated on the control of multimegawatt plasmas during long duration, with the steady state as ultimate goal, ultimately involving steady-state operation of a large tokamak ($R = 2.4$ m, $a = 0.8$ m, $I_p = 1.7$ MA). Both lower hybrid and fast waves are used to drive the current. Enhanced performances related to current profile shaping have been intensively studied (Martin 1997). Consequently, the Tore Supra inner vessel is equipped with several water cooled elements which interact with the plasma to allow injection of several megawatts during several tens of seconds. A world record in injected energy was achieved in 1996 with 280 MJ during a 2-min shot. Very long evolution times were then put in evidence in plasma wall interaction physics.

On the basis of these results, an enhancement in the capability to handle large input powers and control the particles over long duration is now under way. This is the main motivation for the Composants Internes et Limiteur (CIEL) project, which consists mainly in an upgrading of the first wall components:

1. A new toroidal belt limiter in the lower part of the vessel, made from carbon fiber composite (CFC) brazed on copper tubes. This has been designed to remove continuously up to 15 MW of convective power.
2. A set of pumps to evacuate all types of gas species through the toroidal throat of this limiter, allowing improved density control.
3. A new water-cooled radiation screen to cover as much as possible of the inner vessel, to avoid uncooled parts interacting with the plasma.

The Alcator C-Mod Tokamak. The Alcator C-Mod, located at MIT, Cambridge, Mass., is a high-field, high-particle and high-power-density advanced divertor tokamak. It is one of the only five divertor experiments capable of plasma currents exceeding 1 MA. The primary research goals of the experimental program involve divertor research with reactor grade parameters, critical tests of both empirical and theory-based scaling laws for transport, and rf heating and current drive and their application to advanced tokamak research. Figure 5-29 shows a side view of the machine. Note that Alcator C-Mod is enclosed inside a cryostat so that the magnets can be cooled with liquid nitrogen (LN_2) to nearly 77 K. The liquid is circulated through the magnets and then drains back to a 300-ga sump, where it is recirculated.

The C-Mod vacuum vessel, unlike most other tokamaks, is used as a structural element designed to support the OH and EF coils. Since thick stainless steel construction is required, large currents can therefore flow. However, these currents have been successfully compensated for by the control system combined with an excellent set of magnet diagnostics. Table 5-8 lists some of the machine parameters currently obtained and also those expected over the next few years as upgrades are made to the first-wall hardware and rf systems are added.

KSTAR Tokamak. The KSTAR (Korea Superconducting Tokamak Advanced Research) project (Choi et al. 1979) is the major effort of the Korean National Fusion Program to design, construct, and operate a steady-state-capable superconducting tokamak. The project is led by Korea Basic Science Institute and shared by national laboratories, universities, and industry along with international collaboration. The key design features of KSTAR are major radius 1.8 m, minor radius 0.5 m, toroidal field 3.5 T, plasma current 2 MA, and flexible plasma shaping (elongation 2.0; triangularity 0.8; double-null poloidal divertor). Both the toroidal and the poloidal field magnets are superconducting

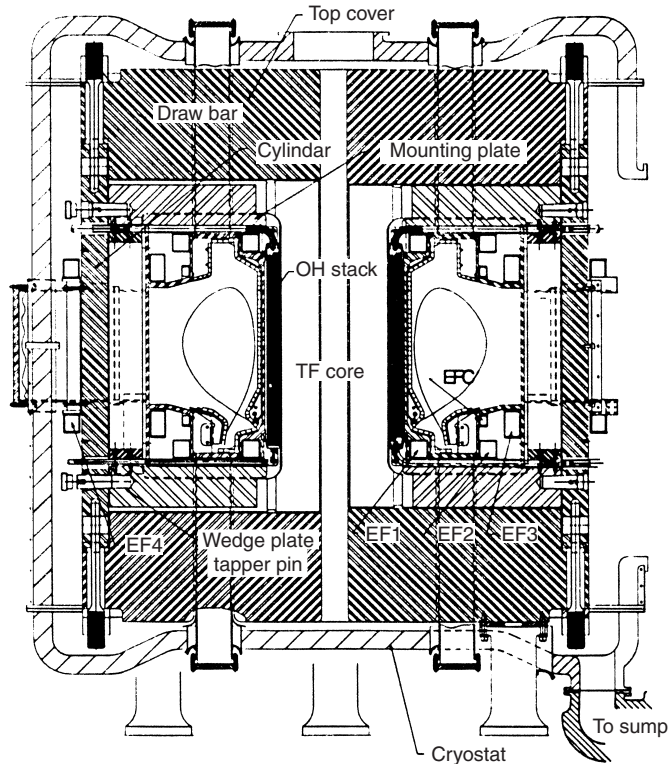


FIGURE 5-29 The Alcator C-Mod tokamak.

TABLE 5-8 Current and Future Alcator C-Mod Parameters

	Current	Future
Toroidal field, T	8	9
Plasma current, MA	1.5	2.5
Density, m^{-3}	$0.2\text{--}15 \times 10^{20}$	—
Elongation	1.8	—
Rf power, MW	4	12
Plasma volume, m^3	1	—

coils. The device is configured for upgrading for up to 300-s operation with noninductive current drive (Table 5-9).

The auxiliary heating and current drive systems consist of neutral beam, ICRF, lower hybrid, and ECRF. Deuterium operation is planned with a full radiation shielding. The KSTAR operational program is staged into two phases: the baseline and the upgrade. In phase I (2002 to 2005), inductively driven 20-s operation with the base-

line auxiliary system (15 MW) is to be conducted, whereas 300-s operation with an upgraded auxiliary system (40 MW) for full noninductive current drive at high-beta regimes is planned for phase II (2006 to 2010).

The International Thermonuclear Experimental Reactor (ITER). The next major step expected in tokamak development has involved a massive international design effort. At the invitation of the director general of the IAEA, representatives of the world's four major fusion programs met in 1987 and developed a detailed proposal for a joint venture called the Conceptual Design Activity (CDA) for the proposed International Thermonuclear Experimental Reactor (ITER). The Director General then invited each interested party to cooperate in the CDA in accordance with the terms of reference developed at the meeting.

TABLE 5-9 Power and Current Drive Plus Projected Operation of KSTAR

Parameter	Baseline	Upgrade
Pulse length, s	20	300
Plasma heating power MW		
Neutral beam	8	24
Ion cyclotron	6	12
Lower hybrid	1.5	4.5
Electron cyclotron	0.5	TBD
Peak DD neutron source rate, s ⁻¹	3.5×10^{16}	
Annual deuterium operating time, s	20,000	
Total number of pulses expected	50,000	

The ITER CDA, under the auspices of the IAEA, began in April 1988 and was successfully completed in December 1990. This work included two phases, the definition phase and the design phase. In 1988, the first phase produced a concept with a consistent set of technical characteristics and preliminary plans for coordinated R&D in support of ITER. The design phase produced a conceptual design, a description of site requirements, a preliminary construction schedule and cost estimate, and an ITER R&D plan. At the end of the design phase in 1998, however, global financial conditions forced a delay in the decision to proceed with construction of ITER. A reduced-size study produced a design for a smaller version at about half the cost.

The ITER design is based on a scientific knowledge base derived from the operation of dozens of tokamaks worldwide over the past decade. It also relies on the technical know-how flowing from the extensive fusion technology R&D programs of the four parties. The ITER tokamak is shown in Fig. 5-30. The principal parameters of the reduced-size version are listed in Table 5-10.

The main characteristics and parameters of ITER follow from its technical objectives which were derived to meet well-defined programmatic goals. Ignition requirements set the value of plasma current. Extended burn favors superconducting coil systems. The design goals for the first wall fluxes and fluence both dictate approximately the same minimum shield thickness. These requirements, combined with considerations of plasma stability, impurity control, and current drive, define the general features and approximate size of ITER. Nevertheless, within the freedom allowed by the technical objectives, the design philosophy has been to control size and minimize cost. Moreover, safety considerations are an integral part of the design activities.

The CDA defined both physics and technology R&D conducted in the various laboratories of the parties designed to validate the technical basis for the design. The physics R&D studies placed emphasis on reducing uncertainties in the divertor performance and energy confinement, extending the burn duration with noninductive current drive, and avoiding disruptions. The main categories of the technology R&D activities involved superconducting magnets, plasma-facing components, nuclear blankets and shields, remote maintenance, fueling systems, and heating and current drive systems.

The EDA phase lasted for 6 years and culminated in a well-documented basis for a decision on construction and siting. Construction would require about 8 years, followed by two "phases" of operation (physics and technology) totaling about 18 years, after which decommissioning would commence.

The EDA work involved about 1200 professional worker years with a team whose strength peaked at about 180 professionals. The cost of this design work, including full overhead and support personnel, was about \$250 million.

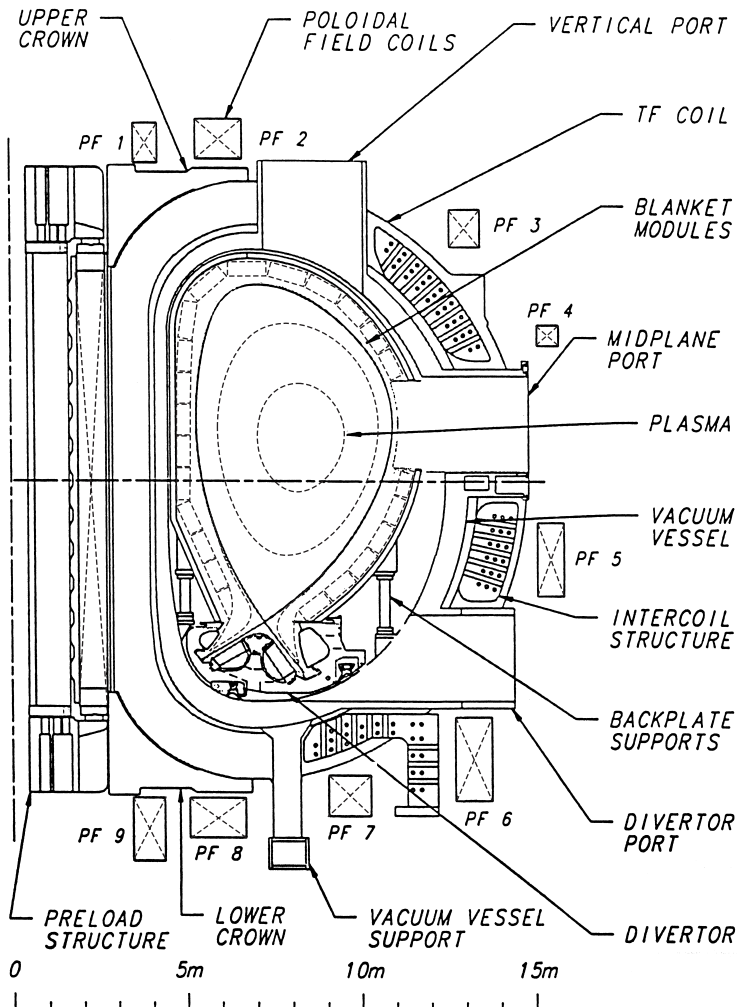
The professional workerpower needed during the construction activities has been estimated to be about 2900 worker-years, primarily at the construction site. The actual construction cost of ITER has been estimated to be about \$4.9 billion. During the construction phase, technology R&D would be needed (estimated to cost about \$300 million). The cost of physics R&D directly attributable to ITER should be added to these costs, but this is difficult to separate out from generic fusion R&D in progress at the same time.

A central team of 300 professionals would be needed during the operation activities, including physicists and engineers, to operate the machine and supervise the testing program. The resulting

TABLE 5-10 Main Parameters and Dimensions of ITER

Total fusion power, MW	500–700
Neutron wall loading, MW/m ²	0.7
Plasma inductive burn time, s	300
Plasma major radius, m	6.2
Plasma minor radius, m	2.0
Plasma current I_p , MA	16
Vertical elongation at 95% flux surface	1.7
Triangularity at 95% flux surface	0.32
Safety factor at 95% flux surface	3
Toroidal field at 6.2 m radius, T	5.3
Auxiliary heating power, MW	73

Source: ITER (U.S.) Home Team Group. *ITER Final Design Report*, IC-33, Project Office, University of California, San Diego, 1998.

**FIGURE 5-30** Cross section of the ITER tokamak.

preliminary operating cost estimates amount to about \$270 million per year. The cost of the experimental part of the ITER program would be in addition to this basic operational cost. Decommissioning costs have not been estimated.

In conclusion, the ITER project has been a pioneering program in both technology and international collaboration. The goal for ITER has been to provide the basis for a demonstration reactor intended to open the way to commercialization of fusion power through the “tokamak route.” Proponents of alternative magnetic confinement approaches and inertial confinement fusion (ICF) have been critical of the size and cost of ITER. They propose alternative approaches which they claim are more attractive and faster to develop. However, the extensive worldwide effort on tokamaks combined with the large experimental database accumulated from the many tokamak experiments carried out thus far have provided the tokamak with the lead in this development to date.

ITER construction is scheduled for 2006 with initial operation in 2014. Two sites have been proposed, one in France and the other in Japan.

Alternative Magnetic Fusion Concepts. Table 5-11 summarizes a representative cross section of alternative fusion concepts (AFCs) that, in one way or the other, have been or are being considered for the production of electrical power, chemical process heat, and/or fissile material. Depending on the confinement scheme considered, systems studies of AFCs range from a simple physics analysis, based on Lawson-like criteria, to detailed conceptual designs. With few exceptions, most reactor studies of alternative concepts fall into the less formalized part of this spectrum. For this reason, a quantitative intercomparison and ranking is not possible.

The toroidal AFCs summarized in Table 5-11 are classified as quasi-steady-state, long-pulsed (10 to 100 s), and pulsed (~ 1 s). A sampling from each category is given.

National Spherical Torus Experiment (NSTX). The National Spherical Torus Experiment (Fig. 5-31) is designed to prove the scientific principle of the spherical torus (ST) plasma. The ST plasma is nearly spherical in shape; its minor radius is slightly smaller than its major radius, thus giving an aspect ratio close to 1. ST plasmas may have several advantageous features such as a higher pressure for a given magnetic field. Since the fusion power density is proportional to the square of the plasma pressure, the ST is an example of an innovative alternative fusion concept that could lead to smaller and more economical sources of fusion energy.

Princeton Plasma Physics Laboratory (PPPL) leads the project and operates the NSTX facility, located at PPPL. The design and construction of the NSTX is a joint project of PPPL, the Oak Ridge National Laboratory (ORNL), Columbia University, and the University of Washington. The NSTX will have modular components for ease of repair and upgrade.

A national research team will carry out the research program on NSTX, which will cover a broad range of fusion and plasma science topics. The ST plasma possesses features of tokamak and spheromak plasmas, as seen in Fig. 5-32. The magnetic-field line of the ST plasma resembles that of the tokamak plasma at the stable, inboard side and that of the spheromak at the unstable, outboard side. This leads to a number of attractive physics features of interest to fusion energy science, estimated according to our present understanding in toroidal fusion science. These include

- MHD-stable plasmas with very high average toroidal beta (plasma pressure divided by externally applied toroidal field pressure at the major radius) in the range of 25% to 50%.
- Nearly complete (greater than or equal to 90%) alignment of self-driven current profile with the required plasma current.
- Strong magnetic shear and magnetic well ($\sim 30\%$), which help stabilize plasma microinstabilities believed responsible for plasma turbulence and rapid energy loss.
- Increased flow shear (by up to two orders of magnitude beyond tokamaks), which tends to suppress the remaining plasma microinstabilities.
- High plasma dielectric constant that permits a new class of potentially efficient rf-heating and current drive techniques, such as the high-harmonic fast wave and the electron cyclotron wave conversion to electron Bernstein wave.

TABLE 5-11 Summary of Alternative Concepts for Magnetic Fusion

Mirror
Simple mirror
Baseball coil
Yin-yang coil
Field-reversed mirror
Tandem mirror
Quasi-toroidal advanced systems
Advanced toroidal systems
Quasi-steady-state
Spherical tokamak
Stellarator
Helitron
Torsatron
Bumpy torus (EBT)
Toroidal bicuspid (Tormac)
Surface magnetic confinement (Surmac)
Long pulsed
Reversed-field pinch (RFP)
Ohmically heated torus (OHTE)
Ohmically heated tokamak (Riggatron)
Pulsed
Theta-pinch (RTP)
High- β stellarator (HBS)
Belt-shaped screw pinch (BSP)
Compact toroid
Stationary
Spheromak
Field-reversed mirror (FRM)
Triggered-reconnected adiabatically compressed torus (TRACT)
Electron-layer field-reversed mirror (Astron)
Slowly imploding liner (LINUS)
Translating
Spheromak
Field-reversed theta pinch (CTOR)
Moving-ring field-reversed mirror (MRFRM)
Ion-ring compressor
Linear
Steady-state
Multiple-mirror solenoid
Pulsed
Linear theta pinch (LTP)
Laser-heated solenoid (LHS)
Electron-beam heated solenoid (EBHS)
Very dense (fast-pulsed, linear) systems
Fast-imploding linear (FLR)
Dense plasma focus (DPF)
Wall-confined shock-heated reactor (SHR)
Dense Z-pinch (DZP)
Passive liners

- Minimized plasma magnetic flux and helicity content to ease noninductive plasma start-up, including coaxial helicity injection or bootstrap current overdrive.
- Naturally diverted plasma scrape-off layer with large flux expansion (up to 10 or more) and magnetic mirror ratio (up to 4:1).

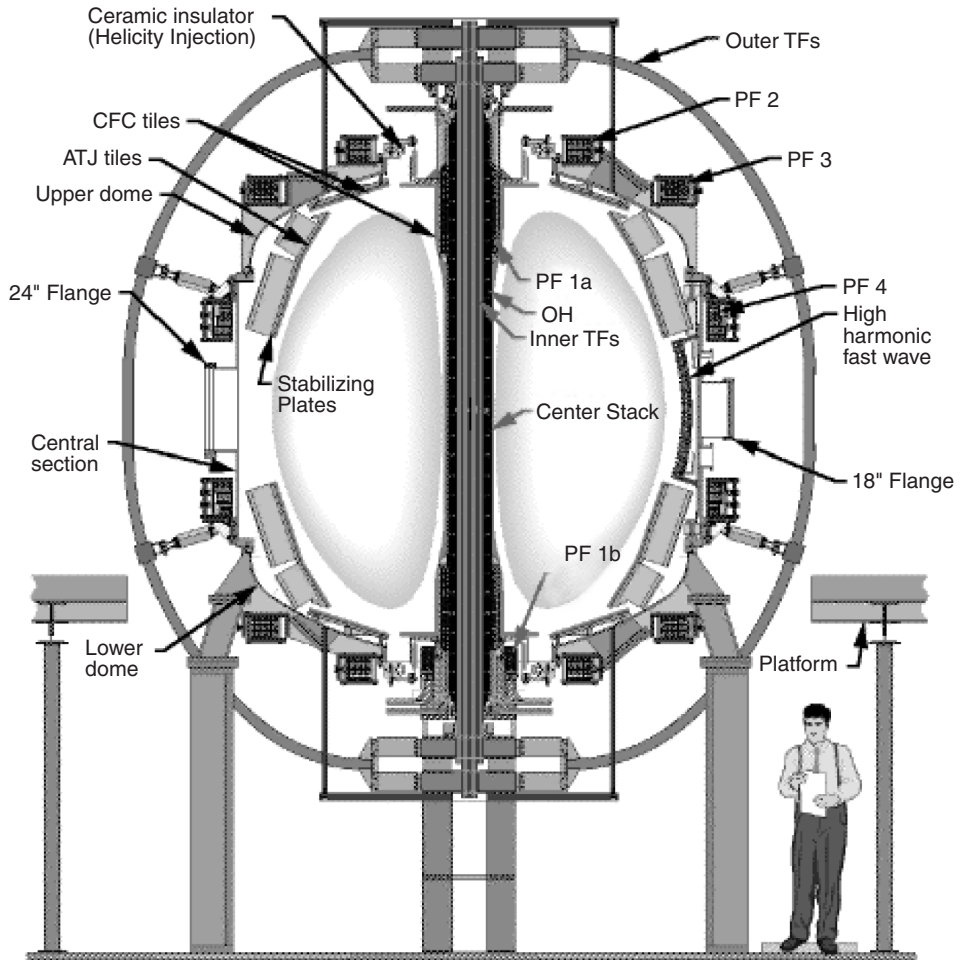


FIGURE 5-31 Cross section of the National Spherical Torus Experiment (NSTX).

These interesting features indicate great opportunities for discovery, innovation, and advancement in fusion and plasma sciences. If verified, the ST plasma will open up new opportunities for attractive applications of magnetic fusion using compact devices.

Figure 5-33 indicates how the ST plasma, given successes in proof-of-principle and proof-of-performance tests, could lend itself to a pathway for developing fusion power.

The NSTX effort grew from successful pioneering experiments conducted worldwide since 1991. These include the START (small tight-aspect-ratio tokamak) at the United Kingdom Atomic Energy Authority-Fusion, UK; the CDX-U (current drive experiment—upgrade) at PPPL; the HIT (helicity injected tokamak) at the University of Washington; the TS-3 (Tokyo Spheromak-3) at the University of Tokyo, Japan; and the SPHEX (spheromak experiment) at the University of Manchester Institute of Science and Technology, UK.

The data from these experiments have shown that the ST concept is ready for proof-of-principle tests. The NSTX is one of the new generation of experiments for this purpose. Other experiments include the MAST (megaampere spherical tokamak) at the United Kingdom Atomic Energy

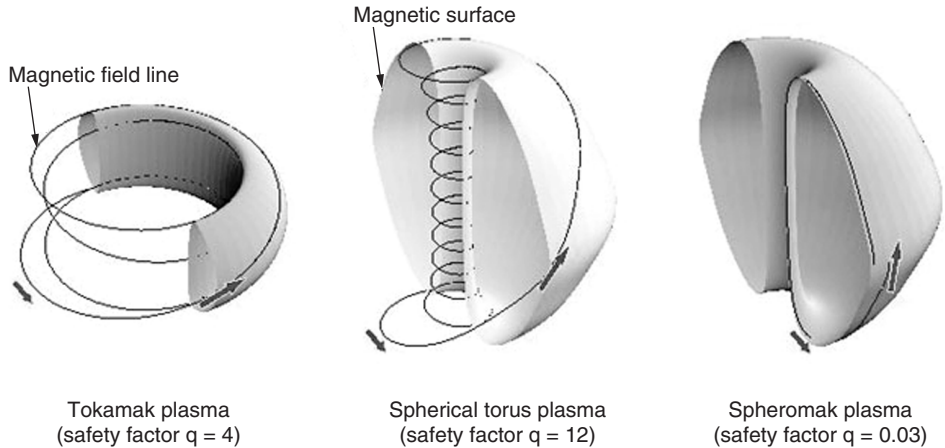


FIGURE 5-32 Spherical torus plasmas combine advantageous features of tokamak and spheromak plasmas.

Authority-Fusion, UK; the GLOBUS-Modified at the Ioffe Physical Technical Institute of St. Petersburg in the Russian Federation; and the Pegasus at the University of Wisconsin. These began experimentation during the second half of 1998.

These experiments will have strong complementary capabilities, and when combined, will permit investigations for plasma currents in the range of 1 MA, stability, toroidal β limits approaching

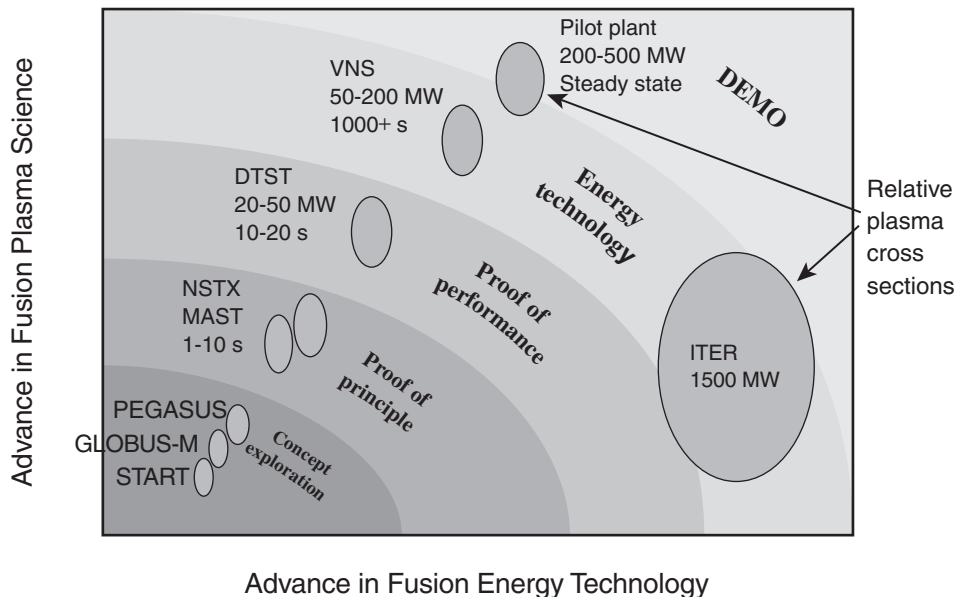


FIGURE 5-33 Schematic diagram of fusion science and technology advances.

50%, bootstrap currents approaching 90% of the plasma current, aspect ratios as small as 1.1, elongation as high as 4, and duration up to 5 s.

Plasma-heating and current-drive techniques to be tested cover neutral-beam injection, a full range of rf waves, and the more exotic noninductive start-up techniques.

The NSTX Research Program will investigate the fusion science principles of the ST plasma, covering

- Noninductive start-up, current sustainment, and profile control
- Confinement and transport
- Pressure limits and self-driven currents
- Scrape-off layer and divertor physics
- Stability and disruption resilience

These principles will be investigated in the scientifically interesting regimes that are relevant to near-term applications such as volume neutron sources (VNS) and fusion pilot and power plants of the future. Thus, the following regimes will be stressed:

- High average toroidal betas (25% to 45%)
- High pressure-gradient-driven current fractions (40% to 90%)
- Fully relaxed, noninductively sustained current profile
- Collisionless plasmas with high temperatures and densities
- Aspect ratios (major radius over minor radius) as small as 1.25 and elongation above 2

The NSTX device (Fig. 5-31) is designed to operate with the baseline parameters in Table 5-12 for plasmas with several forms of divertor and limiter configurations. The NSTX facility shall deliver to the plasma and handle the following power for start-up, heating, and current drive:

- 6 MW of high harmonic fast wave (HHFW) for up to 5 s
- 2 MJ of coaxial helicity injection (CHI) during 5 to 10 ms
- 20 kA of CHI injected current for up to 5 s
- 20 kW of electron-cyclotron heating (ECH) for up to 5 ms

The NSTX facility design will further account for the possibility of adding and handling 400 kW of ECH for up to 100 ms for noninductive start-up and 5 MW of neutral-beam injection (NBI) for up to 5 s for additional heating and current drive. The center stack of NSTX is highly compact to permit a plasma aspect ratio of as low as 1.25. It contains the inner legs of the toroidal field coils, a sole-noid to provide full capability for inductive operation, a pair of poloidal field coils most effective in varying the plasma triangularity, the inboard vessel wall containing these components, the protective

TABLE 5-12 National Spherical Torus Experiment Baseline Parameters

Parameter	Value
Plasma major radius R_0 , m	0.85
Plasma minor radius a , m	0.68
Applied toroidal field B_t , kG	3.0 at R_0
Plasma current I_p , MA	1.0
Plasma elongation κ	Greater than or equal to 2
Plasma triangularity δ	Approximately 0.4
Edge safety factor q	Approximately 10
Plasma pulse flat-top length τ_{pp} , s	About 0.5 (inductive-only operation) about 5.0 (driven-current operation)

graphite tiles, and the interface to the remainder of the device. The center stack is designed to be replaceable without affecting the rest of the device.

The vacuum vessel is 3.6 m in height and accommodates a plasma elongation of up to 2.2 and a triangularity of up to 0.6 for an aspect ratio of 1.25; a higher elongation of about 3 can be obtained for an aspect ratio of 1.4 by operating plasmas with a reduced minor radius.

Auxiliary heating and current drive for current profile control address another essential part of the NSTX mission. High harmonic fast-wave operation is expected to effectively accommodate the high-beta and high-bootstrap-current fraction plasma discharges and fully utilize rf power sources available on-site. A conducting shell, also protected by graphite tiles, will be placed near the outboard plasma edge to enable the investigation of maximized MHD stability limits in plasma beta and bootstrap current.

Mirrors. Mirror-type fusion reactors are characterized by open magnetic-field geometries; that is, magnetic flux passes through a mirror-type device and intersects material walls outside the reaction chamber. In order for such a device to be an adequate container of fusion plasma, it is essential that the end leakage of the plasma be strongly inhibited.

The earliest magnetic mirror configuration used a solenoid with increased magnetic-field strength near its end (Fig. 5-34). In this “simple” mirror, a charged particle travels in a helical orbit around an axially directed magnetic-field line. When traveling into a region of increasing magnetic-field strength, most particles are reflected, that is, their axial motion is stopped and reversed before they can penetrate to the high point of the mirror field. (Some particles with highly directed velocity may still escape. Strong magnets can reduce this leakage, but it cannot be completely eliminated.)

The plasma in a simple mirror is unstable to sideways motion because the magnetic field weakens in directions perpendicular to the coil axis. (The radial weakening of field is evidenced by the concave-inward surface of the magnetic flux bundle.) To solve this stability problem, the simple mirror configuration was replaced by the minimum- B mirror. From the center of this field—produced by a pair of

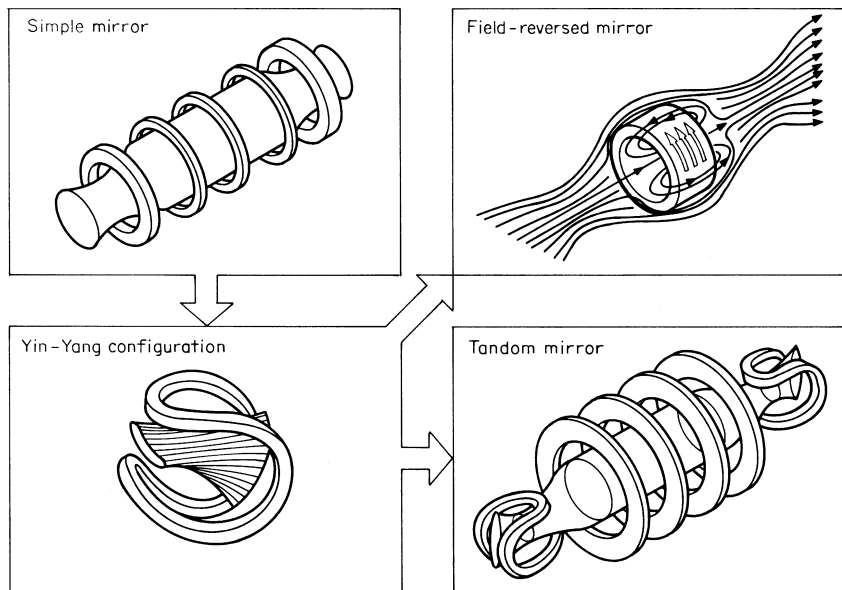


FIGURE 5-34 Evolution of mirror confinement concepts. The field-reversed mirror uses an internal ion ring (created by injection and by diamagnetic currents) to effectively create a torus configuration embedded inside the open mirror field. The tandem mirror uses electromagnetic fields created by ion injection to reduce the leakage from the ends of a solenoidal field.

solenoids and Ioffe bars, a baseball coil (shown in Fig. 5-34), or yin-yang coil—the field strength increases in all directions. However, by 1975, it was concluded from conceptual design studies of mirror reactors based on the minimum- B geometry that end losses from such a reactor would severely limit its plasma Q (fusion power divided by trapped injected power). The subsequent search for enhanced- Q mirror machines led to two new concepts: the field-reversed mirror and the tandem mirror.

In a field-reversed mirror (FRM), the confinement of plasma occurs through a toroidal region of closed magnetic-field lines generated by plasma currents in a nearly uniform background field (Fig. 5-34). The FRM offers the exciting possibility of fusion electric power reactors in small sizes. For example, conceptual design studies were carried out for both a multicell reactor producing 75 MW (electric) of net electric power and a single-cell pilot reactor producing ~11 MW (electric). However, following a series of initial experiments at the Lawrence Livermore National Laboratory (LLNL) that failed to build up sufficient plasma current for reversal, most of the LLNL effort shifted to the tandem mirror.

The tandem mirror confinement concept, invented in 1976, received strong study until cancellation of the U.S. program in 1985. The basic concept entails the improved axial confinement of a long cylindrical fusion plasma within a solenoid by means of strong electrostatic potentials at the ends, produced by mirror-confined, end-plug plasmas (Fig. 5-34). An improvement in the tandem mirror involved use of a *thermal barrier*. This concept uses enhancement of the confining electrostatic potential to establish a hotter electron population in the plugs than in the central cell. However, in the normal operation, electron flow between the plugs and central cell presents large temperature differences. Thus, a thermal barrier is necessary to reduce the passing of central-cell electrons into the plug. Basically, the thermal barrier consists of a region of much reduced magnetic field strength, plasma density, and plasma potential. This causes a depressed positive plasma potential which serves as an electrostatic barrier to electrons.

In order to maintain the thermal barrier, however, it is necessary to prevent the filling of the barrier region by thermal ions leaking from the central cell. Several methods of “barrier pumping” have been investigated. In one, the pumping is accomplished by trapped ions undergoing charge exchange interactions with axially directed neutral beams.

A preliminary conceptual design of a power reactor based on the tandem mirror has been carried out, including use of thermal barriers. The D-T fusion plasma is contained in the 56-m-long central cell and produces 1770 MW of fusion power. With $Q \approx 10$, the reactor produces ~500 MW of net electricity. The central cell consists of twenty-eight 2-m-long modules, each containing an annular blanket region, a magnetic shield region, and two niobium-titanium solenoidal magnets. The entire central cell resides in a vacuum trench, which allows the module-to-module seal to be made by an annular metal inflatable cushion. The plug plasmas, contained in the plug yin-yang coils, are each sustained by a low-current, 400-keV neutral beam. A gyrotron tube system is used for microwave heating of the electrons on the plug side of the thermal barrier. Neutral beams on the end wall of the plug vacuum vessel provide charge-exchange pumping of the barrier and fueling of the central cell.

Advanced Toroidal Systems. Various “advanced” toroidal confinement systems are under active study. The spherical tokamak directed earlier represents one approach. Others include the stellarator, (and torsatron) and the helitron.

Stellarator-Torsatron. Unlike the tokamak, the nonaxisymmetric stellarator achieves equilibrium in a toroidal geometry by externally inducing a rotational transform in the confining magnetic field (Fig. 5-35); ideally, no axial currents need to be supported by the toroidal plasma column, as is required in a tokamak, although until very recently all stellarator experiments utilized such currents for ohmic heating. Implementation of a deformation (twist) into a simple toroidal field coil set allows the Torsatron magnetic geometry to be produced while eliminating the helical coil set in favor of a highly modular device (Fig. 5-36). In addition, more optimally oriented coil forces and lower stresses are anticipated for this modular Torsatron approach. These new advances have renewed interest in the reactor extrapolation of the stellarator-Torsatron concept, a renaissance that coincides with experimental success in heating a low-ohmic-current device, the latter being a prerequisite for eventual steady-state reactor operation.

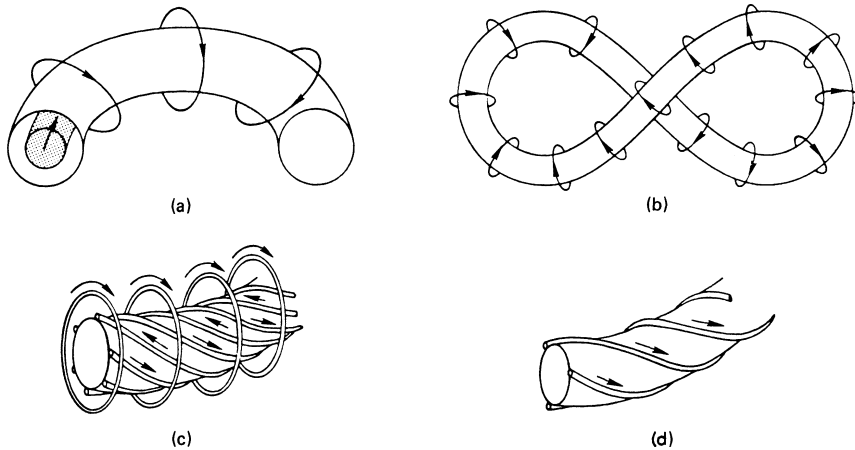


FIGURE 5-35 Toroidal configurations. (a) Tokamak: the main magnetic field is provided by poloidal coils with the poloidal field due mainly to a toroidal current in the plasma. Several external toroidal coils (which are not shown) provide a vertical magnetic field as well as drive the plasma current. (b) Figure-eight stellarator: only one set of solenoidal coils is needed. The rotational transform is generated by the torsion of the magnetic axis. (c) Classical stellarator: a set of $2l$ helical coils with current flowing in opposite directions inside the toroidal field coils provides the rotational transform. (d) Torsatron: a single set of l helical windings provides both the toroidal and poloidal fields. Usually, an additional set of toroidal coils (which are not shown) is necessary to provide an additional vertical field. The standard heliotron configuration is similar to this, but has an additional set of toroidal field coils.

Qualitative advantages that can be invoked for the Torsatron reactor concept include steady-state magnetic fields and burn, operation at ignition or high Q for low recirculating power, impurity and ash removal by means of a magnetic limiter and helical poloidal diverter that occur as a natural consequence of the topology, and no major plasma disruptions that could lead to an intense, local energy dump and no auxiliary position or field-shaping coils and moderate aspect ratio (>10), both of which ease maintenance access.

Stellarator Experiments. Next to tokamaks, the largest worldwide effort in magnetic fusion is focused on stellarator (Boozer et al. 1998, Wakatani 1998). The stellarator concept offers a potentially attractive alternative to the tokamak approach in magnetic confinement fusion. The magnetic field structure, which is required for equilibrium and stability of the confined plasma, is generated by external coil currents only and has an inherent steady-state capability. The rotational transform is generated in the classical approach by helical windings surrounding the plasma vessel. Plasma confinement under net current free operation has been demonstrated in experiments at Garshing, Germany. The net current free operation avoids disruptions.

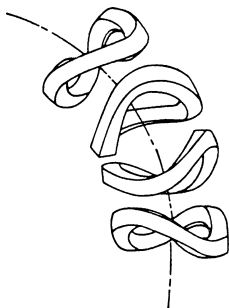


FIGURE 5-36 Coils in a typical modular stellarator.

Two small experiments are located in the United States; HSX at the University of Wisconsin and CAT at Auburn University (major radii of 1.2 and 0.5 m, respectively). Two new major stellarator experiments: LHD and W7-X will provide integrated tests of particular stellarator configurations using superconducting coils. Divertor, transport, and β -limit issues are being studied on CHS in Japan and W7-AS in Germany. TJ-II in Spain (1997) and H-1 in Australia focus on β -limit issues. Stellarator research is also being pursued in Russia and the Ukraine. The theory programs associated with the major stellarators are focused on support for the experiments. Longer-range projects include a better free-boundary package for the MHD stability codes, which is under development at Garching,

TABLE 5-13 Main Parameters of the Last Three Successive Stellarators at IPP

	R , m	a , m	A	B_0 , T	Shear, %	Iota (i)	Coil type
W7-A	2	0.13	15	3.75	2	0–0.6	Helical + toroidal
W7-AS	2	0.18	11	3.0	± 2	0.25–0.6	Modular + planar normal conducting
W7-X	5.5	0.55	10	3.0	15	0.8–1.2	Modular + planar superconducting

Germany. Studies are also starting on the implications of different stellarator configurations for a fusion power plant. A new stellarator, NCSX, is under construction at Princeton.

The Wendelstein 7-X (W7-X) Stellarator. The Wendelstein 7-X Stellarator (W7-X) is the next-step device in the stellarator line of the Institute for Plasma Physics (IPP) in Garching, Germany. A new branch of IPP is in Greifswald, Germany and will house the W7-X.

The stellarator line of IPP concentrates on low-shear devices, which exclude resonances with low rational numbers $n/m = 1/3, 1/2, \dots$ from the confinement region and provide stability by a magnetic well rather than by strong shear. The classical stellarator approach such as W7-A, however, has enhanced neoclassical transport by trapped particles, which is not tolerable in reactor-scale devices. Furthermore, the β limit is too low for an economic power reactor, which requires a β value of typically 4% to 5%. The pressure-driven bootstrap current generates a significant contribution to the total rotational transform in classical stellarators, and thus contradicts the concept that the magnetic parameters can be controlled by external currents alone. These considerations have led to the next-step W7-X device. Some main parameters of the last three successive stellarators at IPP are shown in Table 5-13, where the stepwise development toward W7-X becomes obvious.

The coil and magnetic configuration for W7-X are depicted in Figure 5-37. Fifty nonplanar coils provide the confining magnetic configuration. The flux surfaces show a variation from a strong

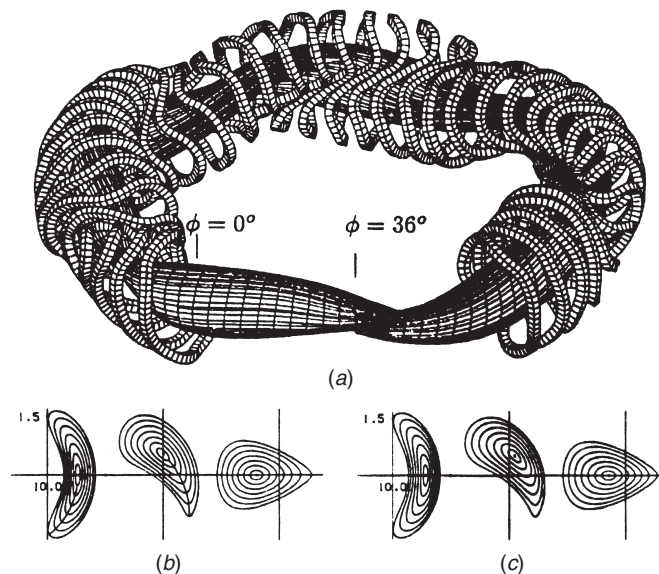


FIGURE 5-37 (a) The coils system of W7-X—50 nonplanar coils provide the confining magnetic configuration; (b) flux surfaces at three positions along one-half magnetic field period for $\beta = 0$; (c) $\beta = 5\%$.

indented elongated shape (bean shape) in the $\phi = 0^\circ$ plane to a triangular shape in the $\phi = 36^\circ$ plane. The standard configuration has a “built-in” profile of the rotational transform with $\tau = 0.86$ on axis and 1.0 at the edge. As is known from W7-AS experiments, the confinement depends strongly on the edge value of the rotational transform. For this reason, experimental flexibility is built into W7-X to allow the variation of the rotational transform and the profile in a wide range. A separate set of planar coils can be independently powered and allows a variation of the rotational transform from 0.75 to 1.01 on axis and 0.83 to 1.25 at the edge. Thus, both low- and high-shear configurations can be verified. The physics goals for W7-X include

1. Demonstration of quasi-steady-state operation in a reactor-relevant plasma parameter regime, with temperatures $T_e < 10$ keV, $T_i = 2$ to 5 keV, and densities $n_e = 0.1$ to $3 \times 10^{20} \text{ m}^{-3}$
2. Demonstration of good plasma confinement to improve the database for reactor extrapolation
3. Demonstration of stable plasma equilibrium at a reactor-relevant plasma β of about 5%
4. Investigation and development of a divertor to control plasma density and impurities

W7-X does not aim at D-T operation, and provisions for remote handling in radioactive environments are not foreseen.

The Large Helical Device (LHD). The LHD at the National Institute for Fusion Science, Nagaya, Japan, is a superconducting (SC) toroidal fusion facility, is a heliotron-type device and has $l = 2$ SC continuous coils and three sets of SC poloidal coils. LHD has a maximum stored energy of 1.6 GJ at 4-T operation (Motojima 1993, 1995). The dumbbell-shaped vacuum chamber is installed in the inside of SC coils and a built-in helical divertor with the double-null structure functions for producing the steady-state plasma. The schematic view and basic parameters of LHD are shown in Fig. 5-38 and listed in Table 5-14. The volume and surface area of the vacuum chamber are 150 m^3 and 850 m^2 , respectively.

To ascertain and realize the steady-state operation of LHD, several important issues concerned with both physics and technology should have been carefully considered and fixed during the design and construction phases. Of interest here are the major operations scenarios of steady-state plasmas based on both physics and technology aspects, especially divertor functions.

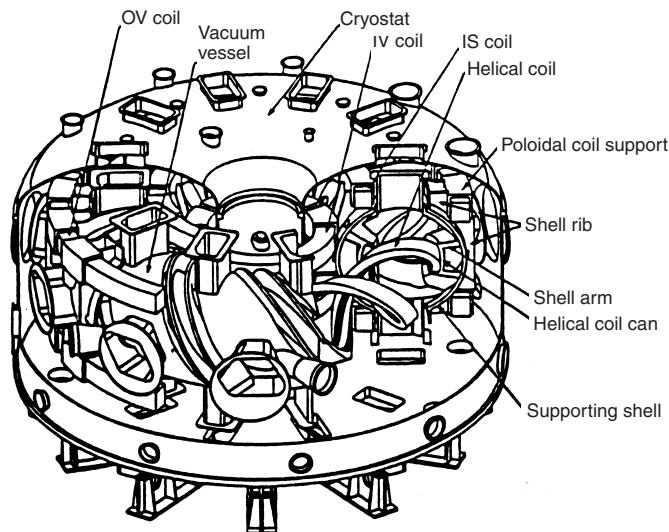


FIGURE 5-38 Schematic view of the large helical device (LHD).

TABLE 5-14 LHD Specifications

Parameter	Value
Major radius, m	3.9
Coil minor radius, m	0.975
Averaged plasma radius l , m	0.5 ~ 0.65
	2, 10
Magnetic field, T	3(4)
Helical coil current, MA	5.85(7.8)
LHe temperature, K	4.4(1.8)
Poloidal coil current	
Inner vertical coil, MA	5.0
Inner shaping coil, MA	-4.5
Outer vertical coil, MA	-4.5
LHe temperature, K	4.5
Plasma volume, m ³	20 ~ 30
Heating power, MW	40
Coil energy, GJ	0.9(1.6)
Refrigeration power, kW	9(~15)

The LHD began operation in fall 1998. The objectives of steady-state experiments are to (1) satisfy the necessary condition for an economical fusion reactor by ensuring self-ignition, possibly taking advantage of the currentless plasma; (2) increase the knowledge of plasma boundary phenomena, plasma-wall interactions, plasma heating (burning), fueling, particle and impurity control, and ash-exhausting and confinement improvement; and (3) advance fusion technology including high-heat-flux components, cooling techniques, and pumping systems.

LHD explores a new parameter regime involving long pulses > 1000 s. The level of the steady (continuous-wavelength [CW]) power is 3 MW at least, with the temperature of more than a few thousand electronvolts in the medium-density region.

The helical divertor is one of the most important design features for control of severe plasma-wall interactions, and for heat and particle fluxes. It is also installed for improving the confinement performance of the plasma.

Elmo Bumpy Torus (EBT). The EBT concept is a toroidal array of simple magnetic mirrors. The promise of a steady-state, high- β reactor that operates at or near D-T ignition emerges from this combination of simple mirrors and toroidal geometry. The creation of an rf-generated, low-density, and energetic electron ring at each position between mirror coils (i.e., midplane location) is needed to stabilize the bulk toroidal plasma against well-known instabilities associated with simple mirror confinement.

Following early experiments on the concept at Oak Ridge National Laboratory, studies of this approach ceased in the early 1990s because of a combination of technical and funding difficulties.

Toroidal Bicuspid (Tormac). Like the tokamak and the stellarator and Torsatron designs, the Tormac is a toroidal device that confines plasma on combined poloidal and toroidal magnetic fields. By opening the outer poloidal field regions, however, the Tormac creates an absolute minimum- B configuration that is MHD-stable for large aspect ratio and plasma β . The resulting toroidal line cusps support plasma on both closed field lines (i.e., high- β bulk plasma) and open field lines; confinement of the latter plasma is enhanced by mirroring effects in the sheath region that separates regions of open and closed field lines. Few experimental studies of this concept have been pursued since an initial series of work in the late 1980s.

Surface Magnetically Confined Systems (Surmac). The Surmac concept represents one example of a general class of multipole configurations in which electrical conductors are arrayed in either a linear or toroidal geometry to create a surface magnetic configuration with low magnetic field in the bulk plasma volume. Figure 5-39 illustrates a toroidal version of this high- β , steady-state system. The Surmac can operate with considerably reduced synchrotron radiation emanating from the bulk plasma, and, therefore, this concept appears to be particularly suitable for confining the high-temperature advanced-fuel plasmas.

Long-Pulsed Toroidal Systems. In terms of power density, relative simplicity, and symbiosis with the basic confinement scheme, ohmic dissipation of toroidal plasma currents represents a very efficient heating scheme. Two long-pulsed toroidal concepts are described that propose ohmic heating as the sole means to obtain an ignited thermonuclear plasma: the Riggatron and the reversed-field pinch (RFP). A variation of the RFP would use an external helical winding to achieve a more controllable rotational transform in a reversed-field state; this concept is called the *ohmically heated toroidal experiment* (OHTE).

The High-Field Ohmically Heated Tokamak (Riggatron Ignitor). Although in principle a tokamak, the Riggatron represents a sufficient change in engineering approach and “conventional” tokamak physics to warrant classification as an alternative concept (Coppi et al. 1992, Carpignano et al. 1995).

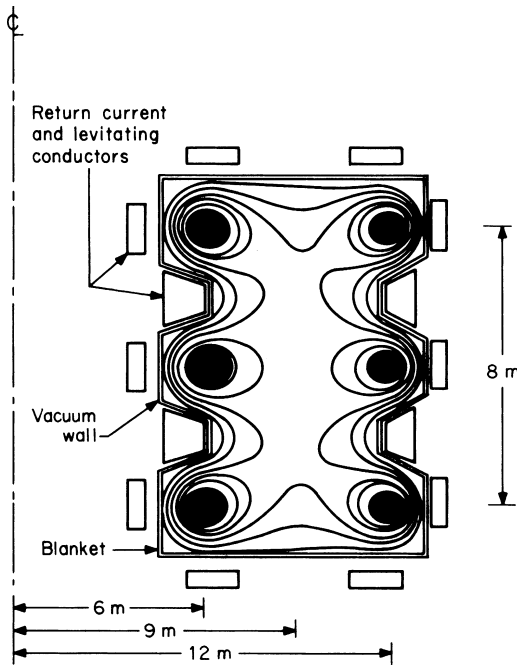


FIGURE 5-39 Schematic representation of the toroidal Surmac plasma and coil configuration.

The combined use of high toroidal current density (8 MA/m^2) and high toroidal field (16 to 20 T) copper coils positioned near the first wall allows net energy production in a relatively short burn period from a high- β , ohmically heated system. The severe thermal-mechanical environment necessarily dictates an engineered short life. The plasma chamber and the D_2O -cooled copper magnets would be small because of the increased plasma density (2 to $3 \times 10^{21} \text{ m}^{-3}$) and high β (0.15 to 0.25). The 6- to 10-MA Riggatron would generate 1 to 2 GW (thermal); the fusion neutron power is recovered in a fixed lithium blanket located outside the magnet system. Recovery of joule and neutron heating in the copper coils is also an essential element of the overall power balance.

The Ignitor experiment involves an advanced compact high-magnetic-field machine, somewhat similar to the Riggatron concept. It employs normal conductor cryogenic magnets with the goal of investigating D-T fusion burning plasmas. The design and construction of the experiment is being carried out by an industrial-university consortium in Italy in collaboration with MIT scientists.

Reversed-Field Pinch (RFP). The RFP is similar to a tokamak in that a toroidal axisymmetrical configuration is used to confine a plasma with toroidal current by a combination of poloidal and toroidal magnetic fields. Using a passive conduction shell, the RFP relaxes inherent constraints on the magnetic-field profiles for a tokamak such that the variation of the magnetic shear need not exhibit a minimum in a region enclosed by a first-wall conducting shell. By removing this constraint, the RFP can operate with a current density that is sufficient for ignition by ohmic heating, an unrestricted aspect ratio, a higher β value, and an appreciably lower magnetic field at the superconducting windings. Consideration has also been given to the possibility of developing a compact reactor based on the RFP concept but using copper coils. The compact RFP reactor would have many operational features similar to those described above for the Riggatron.

Pulsed Toroidal Systems. The early quest on the part of fusion reactor designers to attain the economic advantages of high- β operation simultaneously with the physics advantages of toroidal confinement led to concepts like the reference theta-pinch reactor (RTPR) and the high- β stellarator (HBS). It was generally found that the fast-pulsed nature of the RTPR (i.e., ~ 1 - to 2 - μs shock heating, 30-ms adiabatic compression, ~ 0.5 -s burn time) resulted in technological problems that may outweigh the high- β (>0.8) advantages for that particular system. Additionally, the absence of MHD stability without fast feedback for the particular field configurations then under experimental investigation indicated other reactor-related problems for both the RTP and the HBS, although the latter was eventually proposed for steady-state operation. A more recent variation is the belt-shaped screw pinch reactor (BSPR). While it is still heated by a fast radial implosion, a toroidal bias magnetic field is applied to reduce the final values of β and thereby enhance stability.

Compact Toroids. The generic name *compact toroid* (CT) has recently been applied to the class of toroidal plasma configurations in which no magnetic coil or material walls extend through the torus.

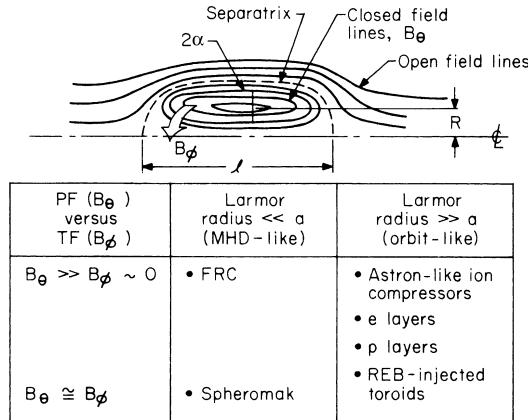


FIGURE 5-40 Diagram and classification of CT configuration where a is the minor radius, R is the major radius, B_θ is the poloidal field, B_ϕ is the toroidal field, and l is the separatrix length.

This closed-field plasmoid configuration is not new, as it was generated by a coaxial plasma gun a few decades ago. Interest in this configuration, as applied to a conceptual fusion reactor, however, began when the spheromak was proposed as a means to retain the developing physics base for tokamaks, while simultaneously shedding certain technological difficulties. Since the spheromak reactor was first proposed, the fusion community has identified the general area and potential of compact toroids; the spheromak is one element of the CT class of plasma configurations. Figure 5-40 summarizes the CT class of devices.

Although representing a great diversity of plasmoid formation, heating, and confinement schemes, the fundamental physics of particle and energy transport and stability and equilibrium are not well known for many subsets of the CT class. Two approaches that involve field reversal with smaller-orbit ions driving the plasma currents—the spheromak and the field-reversed theta pinch—are currently receiving the most study. The theta pinch-reversed field configuration, along with the field-reversed mirror, are normally classified as field-reversed configurations (FRCs).

The FRC and spheromak are closely related. A major difference is the lack of a toroidal field component in the FRC, as its confinement properties derive solely from a poloidal field.

FRC Reactor Studies. Various reactor studies (Momota et al. 1992, Choi et al. 1979, Miley 1988) have been performed to examine the potential advantages of FRCs. Both stationary designs and concepts where the plasma is passed through a linear burn chamber have been considered. The TRACT (triggered-reconnection adiabatically compressed toroid) reactor design for the stationary burn of a field-reversed theta pinch is a good example of these concepts. Table 5-15 also gives typical prototype reactor parameters for this ~1-Hz batch-burn system. Utilizing a longer burn period (~0.5 s) and modest magnetic fields (5.3 T), a hybrid superconducting (dc)/normal (ac) coil system would provide the required flux compression to achieve ignition in a plasmoid with an initial radius of 0.72 m. A first-wall copper coil cancels and subsequently reverses for a few milliseconds the field generated by an exoblanet superconducting coil, during which time a low-temperature

TABLE 5-15 Typical Parameters for TRACT, an FRC Reactor

Parameter	Value
Minor radius, m	0.14
Major radius, m	0.36
Length, m	1.88
Plasma density, $10^{20}/\text{m}^3$	28
Average β	0.77
Magnetic field, T	7
Pulsed energy, MJ	570
Burn time, s	0.5
Thermal power, MW	520
Net power, MWe	130

plasma is created. The superconducting flux is reestablished in two stages: a fast (shock) stage and a slower (adiabatic compression) stage. When the external field induced by the fast shock-heating power supply reaches a peak value, trigger coils are activated and field-line reconnection occurs. The resulting elongated FRC rapidly compresses axially to achieve an equilibrium; this compression provides the bulk of the heating. The resulting ~ 1.5 -m-long plasmoid would attain ignition.

The TRACT approach leads to a relatively small pilot plant of moderate cost that may operate on the basis of near-term technology. A large commercial plant that distributes the power supply costs over several reactor modules benefits since the economy of scale for these systems predicts acceptable direct capital costs. An alternative pilot plant design, SAFFIRE, employed $D-^3He$ fuel and neutral-beam injection for heating plus partial current drive. Pellet injection to maintain the desired radial density profile also contributed significant current drive in SAFFIRE. Reasonable costs for the commercial version would require cost savings by manufacturing of modular components in quantity.

The CTOR system designed at Los Alamos National Laboratory would use a field-reversed theta pinch to produce an FRC plasmoid external to the reactor that is subsequently translated through a linear burn chamber. Like SAFFIRE, in the CTOR design the plasmoid source and compressional heater are removed from the burn chamber to a less hostile environment. To minimize the technological requirements imposed by the plasmoid source and the associated pulsed power, a flared axial compressor would maintain the first-wall magnet coil close to the plasma for stability, while the translating plasmoid is adiabatically compressed to ignition prior to entering the linear burn chamber. Figure 5-41 shows a schematic view of this operation. Translation of the ignited plasmoid in the high-temperature burn chamber allows portions of the conducting shell that have not experienced flux diffusion to be continually "exposed." A nearly steady-state (thermal) operation of the first wall and blanket is possible by adjusting plasmoid speed and injection rate. Superconducting coils are located outside the blanket, conducting shell, and shield to provide a continuous bias field that is compressed between the conducting shell and the plasmoid; gross MHD stability would thereby be provided throughout the burn without requiring feedback stabilization. The plasmoid motion terminates in an end region where expansion directly converts internal plasma energy to electrical energy.

The most extensive FRC reactor design study is for a $D-^3He$ system termed *Artemis*. This study, led by a design team at the National Institute for Fusion Science in Japan, was carried out for the purpose of examining its attractive characteristics and clarifying critical issues for commercial fusion reactors. The $D-^3He$ -fueled FRC fusion reactor *Artemis* consists of a formation chamber, a burning chamber, and a pair of direct-energy converters, all of which are connected by magnetic lines of force (Fig. 5-42). An FRC plasma is produced at start-up by the conventional reverse-biased fast theta-pinch method in the formation chamber and then translated to the burning chamber.

A combination of deuterium NBI, fueling, and a slow magnetic compression in the burning chamber brings the volume, the temperature, and the number density of the plasma up to those for the $D-^3He$ burning state. Most of the $D-^3He$ fusion energy is carried by charged particles along the lines of force, which connect to a pair of direct-energy converters. A smaller fraction of the fusion energy is carried by neutrons and photons to the first wall of the burning chamber. This energy is converted to electricity by turbine generators.

The formation chamber is arranged symmetrically so as to reduce error fields. A fast-rising theta-pinch discharge with a one-turn voltage of 400 kV in the filling gas pressure of 0.05 Pa and bias field

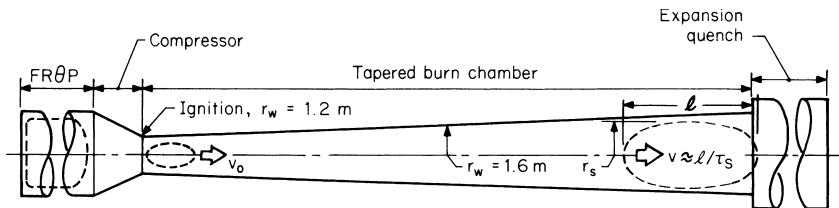


FIGURE 5-41 Schematic layout of a CTOR that would translate at an initial velocity v_0 an FRC down a linear burn chamber of length l . The FRC would be formed externally to the reactor by an FRθP plasmoid source.

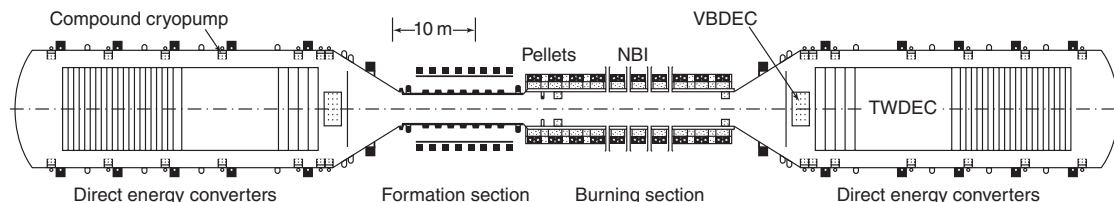


FIGURE 5-42 The D-³He FRC reactor Artemis, composed of a formation chamber, a burning chamber, and direct-energy converters. The venetian blind converter (VBDEC) handles the lower-energy background plasma and alpha particles, while the traveling-wave unit (TWDEC) handles the energetic protons.

of 0.035 T produces an initial FRC plasma (Table 5-16 column a). The plasma is then translated to the burning chamber because of an unbalanced cusp magnetic field. During the translation, the FRC conserves particle numbers of respective species, the total energy, the trapped flux, and its “intelligence.” Plasma parameters obtained at the burning chamber are listed in Table 5-16, column b.

In the burning chamber, the FRC plasma is heated by means of energetic deuterium NBI with a maximum power of 100 MW. A slow magnetic compression is also applied. The injected particles form an ion beam current that serves as the seed current needed to sustain or increase the trapped magnetic flux of an FRC. The plasma evolves in its volume, the density, and the temperature, forming a burning plasma. During the evolution, the ratio of the current carried by the energetic beam particles to the total plasma current is high enough to stabilize an FRC against macroscopic modes. The set of plasma parameters obtained in this way is tabulated in Table 5-16, column c.

In a D-³He burning FRC plasma, the current drive due to the preferential trapping of fusion protons, assisted by a small amount of external beam injection, is sufficient to sustain an equilibrium in a steady burning state.

The concept of a traveling-wave direct-energy converter (TWDEC) was developed to extract the fusion energy carried by 14.7-MeV fusion protons. The technologies needed for this development are conventional. The lifetimes of existing structural ferritic steels are as long as the total reactor life of the Artemis, and no development of new materials is necessary. Fuel injection is accomplished by oscillating the FRC axially to “engulf” fuel pellets dropped into its path. Operation of large high-field magnetic coils is not required. Thus, the bases of technologies employed for Artemis design are conventional and have considerable technological flexibility that enables the design to accommodate a range of unknown parameters.

The estimated cost of electricity (COE) from Artemis is ~30 mill/kWh, which is low compared with a tokamak reactor or even an LWR. The COE depends only weakly on the fuel cost; thus, uncertainties in the cost of ³He are not crucial. The high energy-conversion efficiency (~62% overall) obtained with the TWDEC and the modest cost of that unit are key factors in the competitive COE

TABLE 5-16 Plasma Parameters at Various “Stations”

	Phase		
	a	b	c
Plasma radius, m	0.7	1.0	1.12
Plasma length, m	4.8	14.8	17.0
Plasma temperature, keV	1.0	1.0	87.5
Electron density ($\times 10^{20}/\text{m}^3$)	4.1	0.62	6.6
Trapped flux, Wb	0.086	0.086	3.66
External magnetic field, T	0.56	0.22	6.7
<i>s</i> value	5.9	2.7	9.2
Plasma beta value	0.92	0.83	0.90

Key: a—after formation of the FRC; b—after translation of the FRC; c—steady burning state.

obtained for Artemis. Low radioactivity, low tritium yields, and the inherent safety of the power plant should be readily acceptable socially and ecologically, representing an important advantage of this type of power plant.

Studies on issues such as stabilization of macroscopic modes by means of energetic particles must still be carried out, but the general characteristics of a D-³He-fueled FRC fusion power plant offer a very attractive prospect for energy development for the twenty-first century.

A key issue in the use of a D-³He reactor such as Artemis is the source of ³He. In the design study, lunar mining was assumed and the corresponding cost factored into the COE cited. An alternative approach is to breed ³He, either using a semicatalyzed D-D reactor or from production of excess tritium and subsequent decay to ³He.

FRC Experiments: LSX. Reviews (Haruhiko et al. 1995, Hoffman 1996, Hoffman et al. 1993, Momota et al. 1992) of the research status of the FRC discuss small-scale experiments under way at the following laboratories worldwide: BN (TRINITI, Moscow), TL (TRINITI), TOR (TRINITI), NUCTE-3 (Nihon University), FIX (Osaka University), MRX (PPPL), STX (University of Washington), FIREX (Cornell University), and rotamak (Flinders University). Notable achievements include formation by a theta pinch and by counterhelicity injection, stabilization of the rotational mode, detection of global internal modes, tilting-mode theory, translation and acceleration, identification of transport anomalies, and reduction of convective thermal losses. Immediate issues include the need to demonstrate a long-lived, high-quality FRC, improved theories for global stability and kinetic physics, the development of a reactor-relevant start-up method and current drive, and a better understanding of the plasma energy-transport mechanism. Additional long-term issues include studies of flow stabilization and kinetic stabilization effects, rotating-magnetic-field current drive, and the concept of a traveling-wave direct-energy converter (TWDEC). FRC experiments have undergone a 5-year hiatus in the United States because of the 1990 DOE decision to halt alternative confinement fusion research in favor of concentrated tokamak development. The largest FRC facility, the Large *s* Experiment (LSX), was built during 1986 to 1990, but operated for only 1 year, after which it was converted to studies of plasmoid fueling tokamaks. Its parameters are shown in Table 5-17. A modified version of that facility was reinaugurated in 1998.

The LSX was constructed to explore the stability of FRCs as *s* (device radius/ion gyroradius) was increased beyond the value of 2 (produced in previous small experiments), where stabilizing kinetic effects are theoretically predicted to diminish. Stable, well-confined FRCs were produced with *s* values up to 4, with indications that higher-*s* FRCs would be stable if they could be successfully formed using the theta-pinch techniques. However, before the LSX program was halted, it became clear that theta-pinch formation was limited to flux values in the tens of milliwebers range, while webers of flux are required for a reactor. Thus, flux buildup represents a key area for future research.

The particle-confinement times measured on LSX and other FRC experiments indicate a strong correlation with the parameter $rs/\sqrt{\rho_i}$, where *rs* is the separatrix radius (see Fig. 5-40) and ρ_i is the gyroradius. The general form implies a confinement time scaling ($\tau_N = x_s a^2/D_\perp$) with a basic particle diffusivity [$D_\perp = 2/n^{1/2}$ (10^{21} m²) m²/s], where *a* is the minor radius of the toroid (Fig. 5-40) and *n* is the averaged plasma density. The factor x_s occurs here as a result of the increase in flux (lower beta) for a given radius FRC, and the dependence on density just reflects the $\rho_i \propto 1/n^{1/2}$ relationship for a high β plasma.

This empirical scaling is unfavorable for low-density operation, and thus low-power, steady-state operation. However, there is evidence from recent Japanese experiments (Haruhiko et al. 1995) that an improved low-density confinement mode may exist. Confinement may also improve when current sustainment techniques are applied. Then, the toroidal plasma current is carried either by large-orbit ions or gyrocenter drifts, rather than by simple diamagnetism.

TABLE 5-17 LSX Parameters

Property	LSX experiment
Coil diameter/length, m	0.9/5
Separatrix diameter/length, m	0.45/4
$x_s/\langle\beta\rangle$	0.5/0.87
External magnetic field, T	0.4
Poloidal flux, Wb	0.010
Plasma density, m ⁻³	1.2×10^{21}
Plasma temperature, keV	0.15
Ion gyroradius, cm	0.44
Kinetic parameter <i>s</i>	4

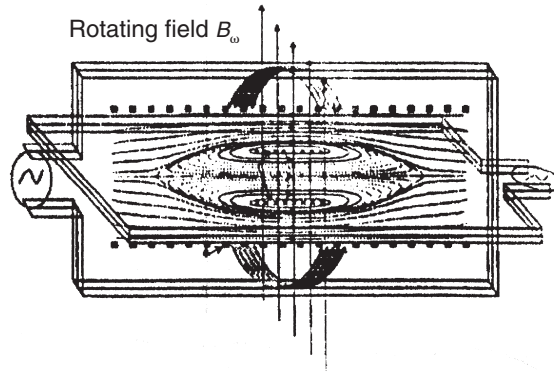


FIGURE 5-43 RMF current-drive schematic.

In order to build a steady-state FRC reactor it is essential to develop some form of flux buildup and current drive. The Artemis reactor design study described earlier assumed neutral beams resulting in large ion orbits (Momota et al. 1992). Large-orbit ions are theoretically effective in stabilizing compact toroids, but the technology is expensive and the large azimuthal ion-ring momentum may introduce ring-plasma instabilities. A simpler, but unproven current drive method for hot plasmas would be to directly drive the electrons using rotating magnetic fields (RMFs). Such a technique, sketched on Fig. 5-43, has been demonstrated on small-scale rotomak experiments.

The two antenna coils in Figure 5-43 are driven 90° out of phase to produce a rotating field $B_r = B_\omega \cos(\omega t - \theta)$, $B_\theta = B_\omega \sin(\omega t - \theta)$, $E_z = r\omega B_r$. If $\omega_{ci} < \omega \ll \omega_{ce}$, then the electrons, but not the ions, will be dragged along with the rotating field. Here ω_{ci} and ω_{ce} are the ion- and electron-cyclotron frequencies, respectively. The perpendicular B_ω field will prevent an axial current from flowing and shielding the plasma interior from the rotating field.

Spheromak Reactor Concepts. The spheromak lends itself to a reactor embodiment (Fowler et al. 1994, Hagenson and Krakowski 1985, Hooper and Fowler 1996, and Moir 1996) that shares many features with the FRC. However, fewer specific studies have been reported for the spheromak. Still it has been shown to potentially offer a comparatively simple fusion reactor with relatively low cost of electricity. The magnetic fields are generated primarily by plasma currents except for a vertical field, required to support the hoop stress, which is generated by a set of purely solenoidal coils. Current is driven by an external coaxial gun that generates linked toroidal and poloidal magnetic fluxes (helicity) that drive current in the core of the spheromak through a magnetic dynamo. The plasma-core current may return to the gun or be collected at the other end of the magnetic separatrix. It is anticipated that the ohmic heating from the plasma current will be sufficiently strong such that no auxiliary heating source will be required to reach reactor conditions.

Figure 5-44 shows the proposed magnetic-field configuration for a spheromak reactor. The fusion plasma is confined within a magnetic separatrix with toroidal, magnetic flux surfaces; thus, the geometry shown has two X points. Helicity injection is from a coaxial, electrostatic gun shown located above the upper X point; power flowing from the confinement region is diverted into the upper and lower coaxial regions, which act as divertors. It may be possible to unbalance the magnetic configuration so that most of the energy and particles flow to the lower region, thus separating the divertor from the gun, and potentially improving the power and particle handling characteristics of the reactor.

The blanket, located in the annular region surrounding the plasma, is envisioned to consist of liquid lithium or lithium salts, perhaps flowing vertically between the wall and the plasma; solid or other configurations are also possible. If ohmic ignition occurs, the spheromak will allow a blanket without radial port penetrations. Thus, it has been suggested that the simplicity of the configuration may make a liquid/NaK "potboiler" type of reactor possible.

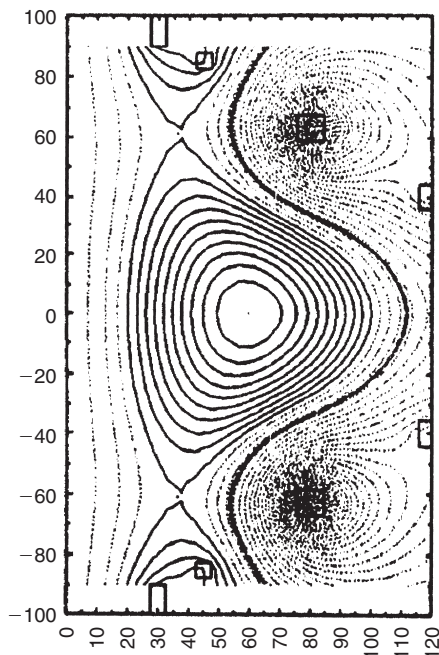


FIGURE 5-44 Flux surface geometry for the spheromak reactor concept. Helicity is injected by flowing current along the open flux surfaces from the upper to the lower divertor regions.

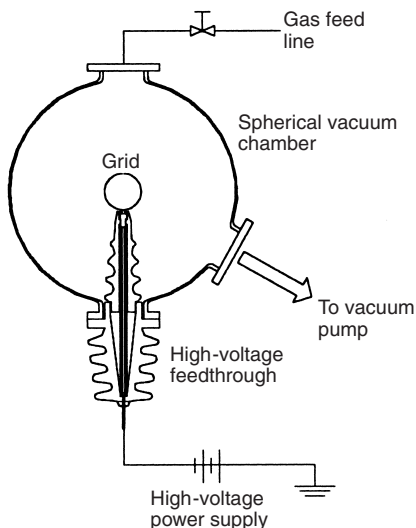


FIGURE 5-45 Cross section showing construction of the single-grid spherical IEC.

5.4.9 Inertial Electrostatic Confinement

Instead of using magnetic fields for confinement, electrostatic fields can be employed (Miley 1997, Kulcinski et al. 1997, Miley et al. 1997, Sved 1997, Gu and Miley 1995, Miley et al. 1998). This requires a dynamic plasma, for instance, one where the inertia of high-energy ions becomes significant since, as proved by Earnshaw, pure electrostatic fields cannot be used alone to confine a plasma. Thus, electrostatic fields have generally been viewed as a way to “plug” plasma leakage routes in magnetic confinement systems, which has been widely considered (e.g., see the preceding discussion of the tandem mirror), and studies of electrostatic confinement have been limited. The best known concept, termed *inertial electrostatic confinement* (IEC), uses the inertia associated with recirculating ions to overcome Earnshaw’s theorem. This concept traces back to Philo Farnsworth, the U.S. inventor of electronic television, who proposed and patented aspects of the concept in the 1960s. While some continuing research has been done ever since then, the first aggressive small-scale experimental projects were set up at the University of Illinois, Los Alamos Scientific Laboratory, University of Wisconsin, MC² Corporation, and the University of Kyoto.

A significant result of these efforts is that Daimler-Benz Aerospace Corporation initiated commercial production of a version of the University of Illinois unit. The commercial IEC unit is for use as a portable 2.5-MeV neutron source (D-D fusion) for industrial neutron activation analysis (NAA) and neutron tomography. While operating with a net power input well below energy breakeven ($Q \sim 10^{-6}$), this development represents the first commercial use of a confined fusing plasma.

The University of Illinois experiment is representative of present IEC research and will be briefly described here. For simplicity, it replaces the ion guns used in early Farnsworth-type experiments with a grid-produced plasma discharge. Operation employs the “star” mode, where ion beams formed in the discharge pass through grid openings, minimizing grid sputtering and erosion.

Both spherical and cylindrical versions have been developed. In the spherical design, illustrated in Fig. 5-45, the transparent grid biased at -60 kV acts as a cathode relative to the grounded vacuum vessel wall.

When a deuterium gas (or deuterium plus tritium or ^3He mixtures) discharge is used, D^+ ions are extracted from the plasma by the cathode grid, accelerated, and focused in the center where D-D fusion reactions occur. The grid provides recirculation of the D^+ ions, increasing the power efficiency. At high currents, a

potential structure develops in the nonneutral plasma, creating virtual electrodes that further enhance ion containment and recirculation (Fig. 5-46).

Steady-state spherical IEC units typically produce 10^7 D-D n/s, while advanced pulsed versions extend to 10^{11} neutrons per second (n/s) (10^{13} n/s for DT).

The cylindrical version of the IEC (Fig. 5-47) uses a hollow electrode design where ions recirculate between two outer anodes producing a line-like neutron source within the hollow cathode region. This unit is most useful for applications requiring broad-area extended sources, such as luggage inspection, and NAA of materials, such as ores, on a conveyor belt.

Neutron yields from the cylindrical design are comparable to those from the spherical IEC, but the lack of three-dimensional virtual electrode formation limits its ultimate fusion rate.

The unique feature of the IEC is that it develops a fusion-grade plasma in a small volume (a few cubic centimeters) while avoiding bulky magnets, injectors, and the like. In addition, the star mode creates ion beams that reduce grid bombardment, which, combined with use of a “damage free” plasma target (the dense core of Fig. 5-46) provides long-lifetime units. Thus, the commercial IEC NAA units provide a first step in what could be a progression of uses for fusion-grade plasmas prior to actual fusion power plants.

In addition to NAA, the IEC has been proposed as a tunable x-ray source and as a plasma thruster for space applications. These applications generally require higher reaction rates, and present research is focused on that objective, largely via use of higher ion currents. If successful, the scale-up to a near-breakeven unit could be considered for neutron-irradiation applications such as neutron-damage studies. Development issues include the stability of the potential structure illustrated in Fig. 5-46 at high currents and technology such as grid survivability. Grid scale-up requires development of very efficient active cooling methods, or use of an alternative approach to eliminate grids. For example, a recent concept proposed by R. W. Bussard replaces the grid with a unique high-order cusp magnetic field.

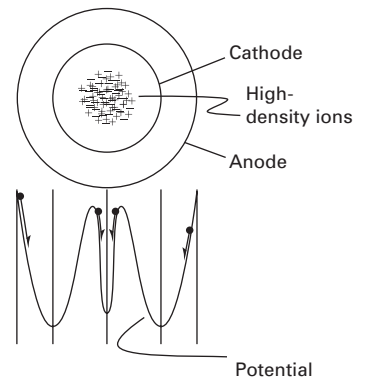


FIGURE 5-46 The potential structure in the dense plasma core of an IEC. The central “well” traps ions, providing efficient local recirculation.

5.4.9 Inertial Fusion Energy and Concepts

In the inertial confinement scheme, the surface of a small spherical target of a solid material containing deuterium-tritium fuel is symmetrically illuminated by very high energy laser or charged particle

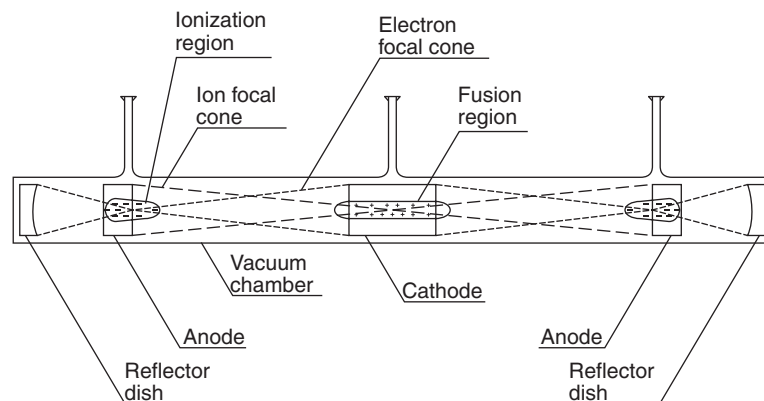


FIGURE 5-47 Cross section of the cylindrical version of the IEC. The ion-electron optics are designed to provide a central beam through the hollow cathode.

beams. The density of the nuclei in the target prior to illumination is typically $\sim 10^{22}$ ions/cm³. A thermal compression, caused by ablation as the beam energy impinges on the target, causes a further increase in density of up to 10^4 times, producing a density at the center of the pellet approaching 10^{26} ions/cm³. At this density, a Lawson-type breakeven density-confinement time product of 10^{16} s/cm³ can be achieved in a confinement time of only 1 ns (10^{-9} s) or less, which is achieved by the inertia of the fuel itself. If a central pocket of DT gas is left in the center of the target, as shown in Fig. 5-44, it will be adiabatically heated during compression to a temperature of 10^8 °C, initiating a fusion burn. These density-confinement, time-temperature conditions are adequate for energy breakeven. Thus, the process is essentially one of a thermally induced implosion. The total laser energy which must be applied to the target to achieve useful fusion gains is estimated to be of the order of 5 MJ. With a target gain of 100, this would give a yield of 500 MJ. A pulse rate of 2 Hz would produce 1000 MW.

In terms of power plant design, the most important feature of inertial fusion is that the confinement of the plasma is decoupled from the functions of the reaction chamber, wall, and blanket. This is possible because the physics of energy absorption, implosion, ignition, and burn of this plasma are physically separable from the chamber conditions and structural requirements.

In inertial fusion power plant chamber design, the degree of freedom allowed by this separability is exploited by placing fluids (gases or liquids) or fields (magnetic deflection) inside the chamber to protect the wall from ablative material loss due to the energy deposition of the short-range fusion radiation (25% to 35% in soft x-rays and debris) and, in the case of thick liquid-metal walls, to provide protection from radiation damage due to deeply penetrating x-rays and neutrons. Similarly, the final optics of the driver can be protected by placing the final mirrors tens of meters away and protecting them by a gas, or in the case of an ion beam driver, by closing a rotating shutter in front of the ion beam ports to block the plasma debris.

The use of protective fluids, combined with the pulsed nature of the energy release, leads to new engineering constraints and techniques that are radically different from those of magnetic fusion. Understanding and manipulating the dynamic response of fluids and structures become of primary importance, and such factors as new materials development to resist neutron damage take less emphasis.

The HYLIFE System. The most detailed calculations and conceptual design for an inertial fusion power plant were performed at LLNL, in conjunction with a team of university and industrial contractors. The high-yield lithium injection fusion energy (HYLIFE) chamber shown in Fig. 5-48 is designed to operate with yields of a few thousands megajoules at rates of a few hertz. A lithium energy-conversion blanket, consisting of a dense array of 20-cm-diameter jets, is continuously injected into the chamber. This provides an effective blanket thickness of 1 m between the fusion pellet and the inner steel wall. The 14-MeV neutron flux is reduced by a factor of 200 by this lithium blanket, allowing a chamber with a 5-m-radius to operate for 21 full-power years (30 years at 70% availability), at an integrated neutron energy flux of 0.3 MW/m². Since the flux in the wall without lithium protection would be 5.7 MW/m² (including pellet effects), the power density within the reactor vessel is very high, approaching that of a fission reactor. A common low-alloy ferritic steel (2.25 Cr-1 Mo) is used throughout. The tritium-breeding ratio is controllable between 1.0 and 1.7 by adjusting one G/i enrichment in the lithium.

The chamber contains a hexagonal array (50% packing fraction) of 20-cm-diameter lithium jets, which gives an effective shock isolation from the effects of the fusion pulse. Since the hot gas merely blows through the array of jets, this configuration minimizes the wall stress due to the impact of lithium accelerated by high-pressure blowoff gas, caused by x-ray and debris energy deposition. A substantial fraction of the kinetic energy of expansion of fluid, resulting from the neutron absorption, is dissipated in the liquid-liquid interactions among colliding jets. Finally, the enormous surface area of the fluid acts as an effective condensation pump for the lithium vapor.

The HYLIFE power plant requires 16 pumps to inject the liquid lithium into the 5-m-radius chamber. The power necessary for the lithium recirculation pumping is 20 MW (electric), only 1.6% of the gross electric power, if mechanical pumps are used. The laser mirrors are located at the far ends of the containment building, 60 m from the microexplosion. They are protected from the soft x-rays and debris by 0.25 torr of xenon in a 30-m section of the passageway. The neutron flux on the

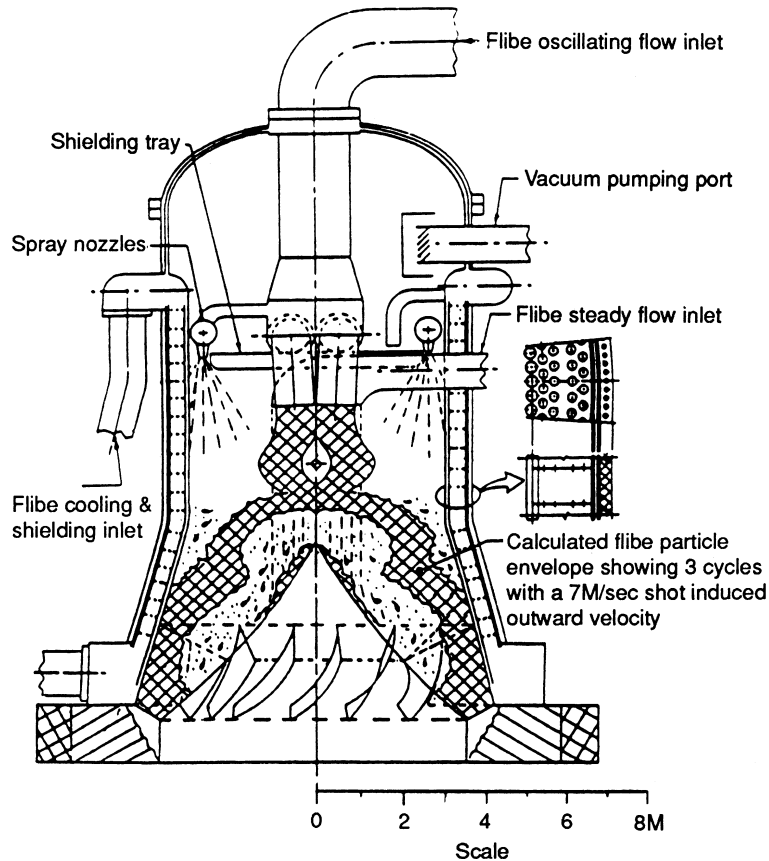


FIGURE 5-48 HYLIFE-II, oscillating flow combines wall protection and driver-beam access.

mirrors is 0.042 MW/m^2 , low enough to assure substrate lifetimes of over a year. The power plant characteristics are summarized in Table 5-18.

HYLIFE-II. The HYLIFE-II reaction chamber, shown in Fig. 5-48, employed an oscillating flow scheme to permit beam injection while still protecting the chamber wall. HYLIFE-II avoids the fire hazard of lithium by using a molten salt composed of fluorine, lithium, and beryllium (Li_2BeF_4) called Flibe. Access for heavy-ion beams is provided.

A high repetition rate is employed in HYLIFE-II, requiring faster reestablishment of the jets after a shot. This is accomplished in part by decreasing the jet fall height and increasing the jet flow velocity. In addition, there is liquid splash that must be forcibly cleared because gravity is too slow at repetition rates higher than 1 Hz. Splash removal is accomplished in the central region by oscillating jet flows.

Flibe is compatible with 316-stainless steel or even better with Hastelloy N at a much higher temperature than lithium (923 vs. 770 K). Its heat-transfer properties, while different from lithium, are still suitable to remove heat and serve to protect the permanent structure from neutron damage and blast. Flibe, unlike lithium, dissociates. However, recombination is rapid enough to avoid affecting condensation; hence this does not limit the repetition rate. If the salt is maintained in a reduced state

TABLE 5-18 Summary of HYLIFE Power Plant Characteristics

System parameters	
Fusion power, MW	2700
Net electric power, MW (electric)	1004
Net system efficiency, %	32
Tritium breeding ratio	1.0 to 1.7
Laser and pellet parameters	
Beam energy, MJ	4.5
Pellet gain (energy multiplication)	400
Yield, MJ	1800
Repetition rate, Hz	1.5
Laser efficiency, %	5
Laser power consumption, MW (electric)	135
Fusion energy gain	20
Fusion chamber	
Radius, m	5
Height, m	8
Material	$2^{1/4}$ Cr-1 Mo
Midplane neutron flux, MW/m ² (with lithium)	0.68
Lithium array geometry	
Number of jets	175
Midplane jet diameter, m	0.2
Midplane packing fraction	0.57
Effective array thickness, m	0.47
Flow parameters	
Midplane jet velocity, m/s	13.3
Array flow rate, m ³ /s	72.2
Total lithium pumping power, MW (electric)	26.0
Total lithium inventory, m ³	1850
Lithium outlet temperature, °C	500
Lithium temperature rise per pulse, °C	18

Source: Krakowski, R. A., Glancy, G. E., and Dabiri, Ali E., The technology of compact fusion reactor concepts. *Nuclear Technology/Fusion*. 1983, vol. 4, p. 342.

by keeping the mole fraction of tritium fluoride (TF) low, corrosion from fluorine compounds formed during the evaporation process should be small. The salt is reduced by contacting it with beryllium or by use of a $\text{CeF}_3/\text{CeF}_4$ buffering agent.

To achieve a high repetition rate and short fall distance with splash clearing, the jet nozzles oscillate horizontally, as shown in Fig. 5-48.

“Cool” Flibe at 873 K is injected in a spray of droplets in the vicinity of the beam paths. This injected spray can provide a large condensation area. Thus, it is not necessary for the explosion itself to form a large number of small droplets for seeding the condensation process as is sometimes assumed.

Practically, all of the tritium gas emitted by exploding targets will be removed by the vacuum pumping system, but almost none of the tritium bred in the Flibe will diffuse out of the Flibe droplets while in the chamber. The reason is that tritium has a low solubility in Flibe so it tries to diffuse out of and not into the droplets. A vacuum disengager has been designed to remove the tritium from the molten salt coolant (see Fig. 5-49).

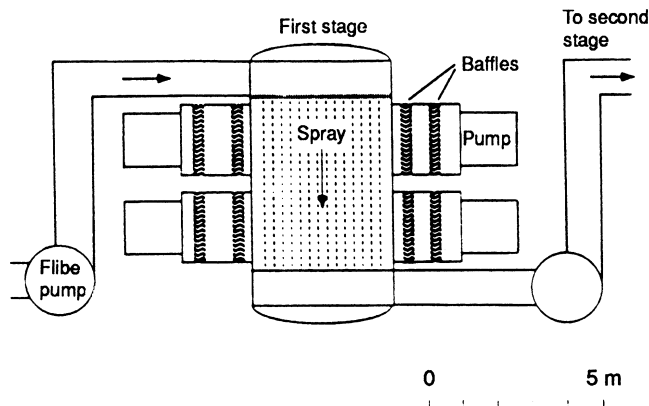


FIGURE 5-49 The vacuum disengager removes tritium from small Flibe droplets falling 5 m.

An outstanding feature of the HYLIFE-II reactor is its favorable safety characteristics. These include

- Off-site dose from severe accident less than 2 Sv (200 rem) for passive safety
- No N-stamp requirement for most components, thus requiring less than 0.25 Sv (25 rem) off-site dose
- Working area dose rate less than 50 mSv/h (5 mrem/h) for a low occupational risk
- Dose from routine atmospheric effluents less than 50 mSv/year (5 mrem/year)

Some of the key issues include verifying splash removal techniques, tritium removal effectiveness, condensation phenomena, heavy-ion accelerator technology, cost reduction, and beam propagation. Most importantly, to be competitive with future coal and light-water-reactor (LWR) nuclear power, the cost of electricity needs to be reduced by a factor of somewhat less than 2.

OFE Inertial Fusion Reactor Studies. There have been a number of ICF reactor studies such as HYLIFE with varying degrees of completeness (Miley and Campbell 1997). Recently, the Office of Fusion Energy (OFE) commissioned two studies to take advantage of the database through 1990, and we will briefly consider them. These studies take 1 GWe as the appropriate output electrical energy (a large power plant).

The OFE reactor studies have considered both heavy-ion and KrF laser drivers. Both approaches to ICF, namely, direct and indirect drive (qualitatively illustrated in Figs. 5-50 and 5-51), are considered. (Other studies are advancing the diode-pumped solid-state laser discussed earlier. A light-ion-beam driver study is also being advanced.) However, with at least thousands of active components, it is difficult to accurately assess the reliability and true costs of these conceived drivers. Figures 5-52 and 5-53 illustrate delivery of the driver energy in the Prometheus-H (heavy ion) and SOMBRERO (KrF laser) concepts that are taken as typical. Prometheus-H has a single accelerator, but 18 beamlets are generated, stored, delivered, and focused from two sides onto the indirectly driven target.

The SOMBRERO layout illustrates some of the issues for laser direct drive. Note that the building has a 55-m radius. The size of the building is dictated by the need to place laser optics far from the neutron source.

The ability to line up and consistently hit a target is a key issue. Figure 5-54 illustrates a delivery concept for the SOMBRERO case. The targets must be delivered accurately and, particularly in the case of indirect drive, they must also be precisely aligned. The OSIRIS reactor chamber concept (Fig. 5-55) has highly attractive features. The walls are wetted and continuously renewing. Liquid Flibe (500 to 600°C) is the wetting agent, the heat-removal agent, and the tritium-breeding agent.

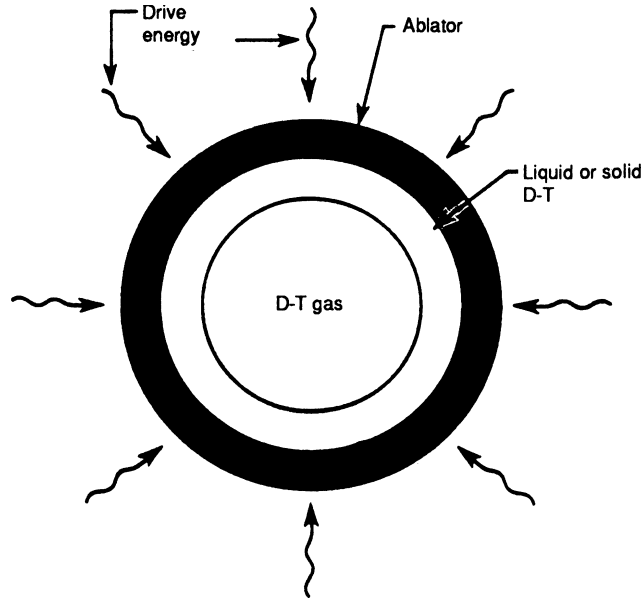


FIGURE 5-50 The basic concept of inertial fusion uses uniformly distributed radiation to drive the compression of a capsule containing deuterium and tritium. Direct drive applies the incident energy directly from a laser or ion beam.

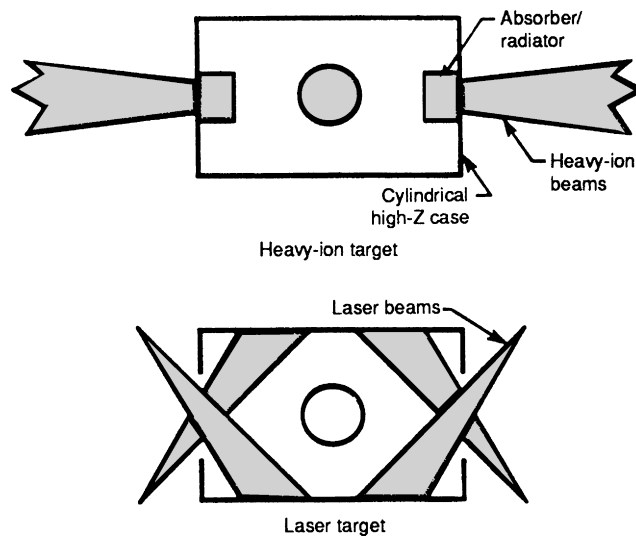


FIGURE 5-51 For indirect-drive targets, the incident ion beam or laser energy is focused onto a material surface where secondary radiation is generated, resulting in uniform radiation incident on the central, spherical capsule.

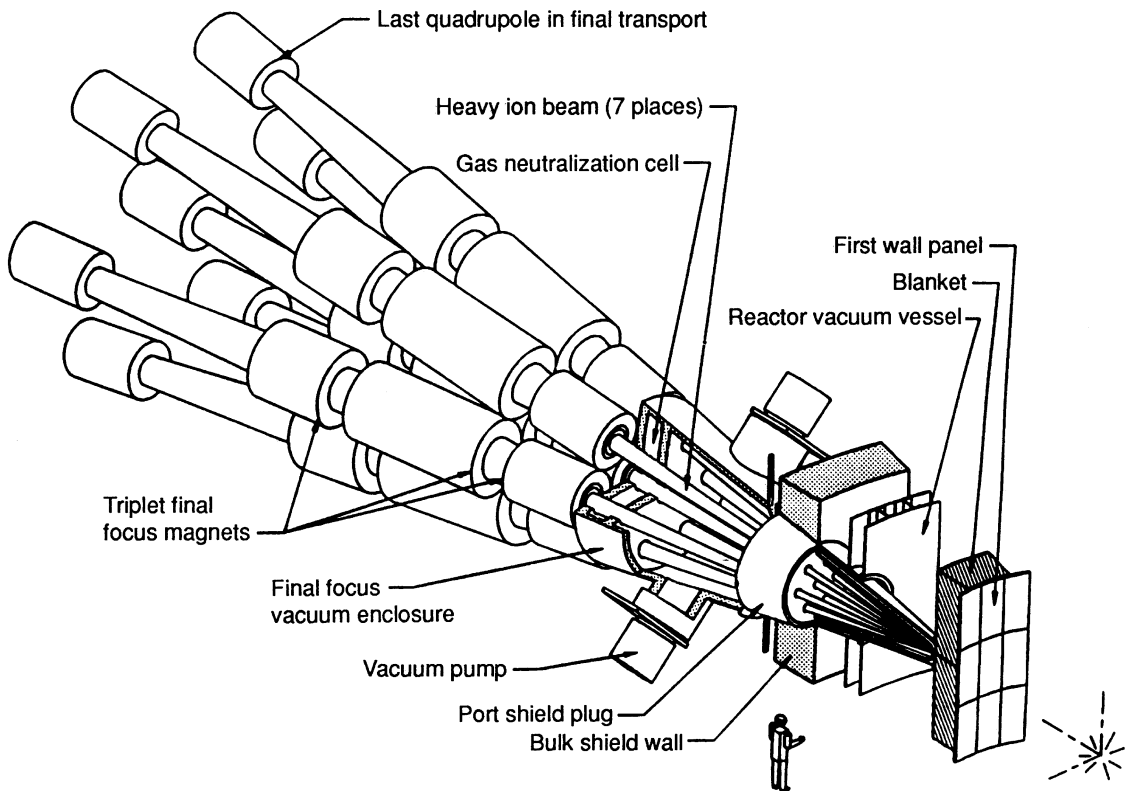


FIGURE 5-52 Side view of final focusing arrangement for one side of heavy-ion-driven Prometheus-H. (McDonnell-Douglas.)

Fast Ignitor. In addition to research on direct- and indirect-drive targets for inertial confinement fusion (ICF), worldwide interest has developed the concept of “fast ignition” proposed in 1994 by researchers at LLNL (Tabak 1994). In this concept, the target is first “hit” by the main driver beam which compresses it to a high density. Almost simultaneously, it would be “hit” by a second beam from a Petawatt laser, which uses chirped-pulse amplification to achieve very high power, with short time pulses through a thousand-fold pulse compression in time (Perry et al. 1996).

When focused onto a solid target, the Petawatt laser pulse can generate a forward-traveling beam of high-energy electrons. The fast-ignitor concept is to use this electron beam to “bore a hole” through the outer region of a precompressed ICF target, igniting a small volume of fusion fuel near the center (equivalent to “spark ignition” by converging shock waves created by a time-shaped laser pulse in normal direct-drive ICF).

Fast ignition can reduce the driver-beam energy by 10 times or more for inertial fusion energy (IFE) by reducing the implosion velocity and energy for a given fuel compression ratio (Logan and Meier 1998). For any type of driver that can deliver the ignition energy fast enough, fast ignition increases the target gain compared to conventional targets. By reducing the fuel compression energy, velocity, and implosion convergence ratio, and by tolerating mixing in the DT core, fast ignition also reduces the degree of symmetry required both on the driver beam illumination as well as on the smoothness of capsule ablaters and cryogenic DT fuel layers. In principle, these advantages of fast ignition can accrue to any type of driver used to compress the fuel, and to any driver type used to ignite the fuel after compression; that is, ion accelerators or lasers might be used for either or both

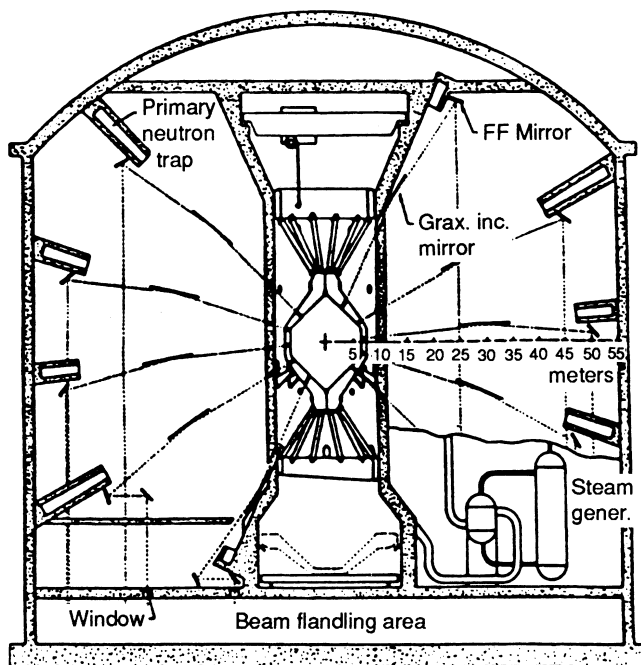


FIGURE 5-53 Side view of full reactor hall for the SOMBRERO KrF laser-driven concept. (W. J. Schafer.)

functions. With high peak beam power and small focal spots, fast ignition is more demanding of the driver than is fuel compression, despite lower-energy input required for ignition. Experiments with a Petawatt laser have demonstrated adequate conversion efficiency of light into hot electrons with the range needed to ignite DT cores in future experiments (Perry et al. 1996). Further experiments are required to determine if the laser light and hot electrons can channel deeply enough into the corona plasma of a preimploded capsule to heat a small portion of a >1000 -fold liquid density compressed DT core to ignition. Ion-driven fast ignition, compared to laser-driven fast ignition, has the advantage of direct (dE/dx) deposition of beam energy to the DT, eliminating inefficiencies for conversion

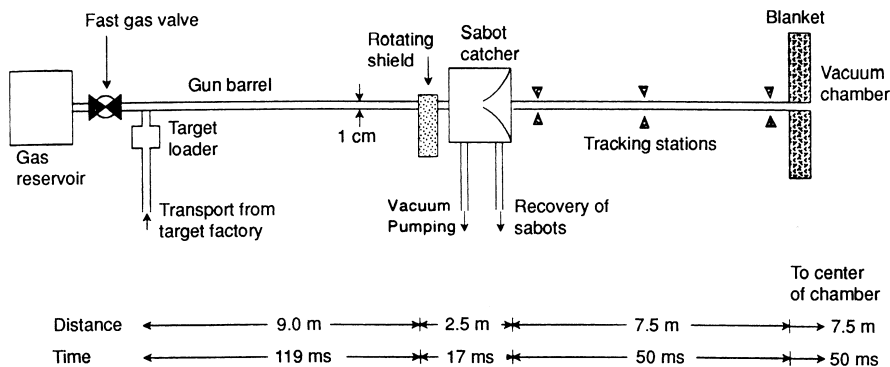


FIGURE 5-54 Target injection system for the SOMBRERO ICF reactor design. (W. J. Schafer.)

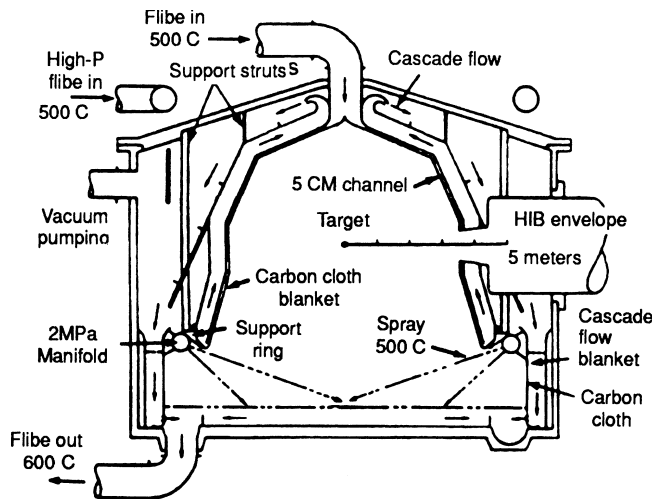


FIGURE 5-55 Side view of OSIRIS, heavy-ion-driven concept. The Flibe flows in through the narrow, 5-cm channel and soaks through the carbon-cloth blankets, providing a self-renewing wetted wall. The Flibe then circulates through the blanket, absorbing the neutrons for thermal energy and tritium breeding. (W. J. Schafer.)

into hot electrons. On the other hand, ion accelerators have not yet been demonstrated that can produce sufficient beam quality for high pulse compression ratios and small focal spot sizes for fast ignition. Accordingly, it is the accelerator and beam focusing, not the target coupling physics, that poses the main challenge to ion-driven fast ignition.

ICF Laser Facilities NIF. The NOVA laser at the Lawrence Livermore National Laboratory (LLNL) in California has served for years as the world's most powerful laser for inertial confinement fusion (ICF) research. It was shut down to make way for the National Ignition Facility (NIF). This facility has as its ultimate goal the demonstration of ignition and high-gain operation of ICF targets. To accomplish this, the NIF is designed to deliver over 2 MJ of 0.35- μm light to indirect-drive targets with a peak power of 500 to 1000 TW in a variety of pulse shapes.

For ICF, ignition is defined where the fusion alphas deposit sufficient energy locally to double the burn temperature in the hot spot of the fuel. This will occur in a DT target if the core is compressed to $\rho r > 0.3 \text{ g/cm}^2$ and a preignition temperature of 5 keV. Then a thermonuclear burn will propagate into the remaining cold fuel, which is nearly isentropically compressed to high density. For a reactor, an energy gain of 100 or more would be desirable.

The National Ignition Facility. The National Ignition Facility (NIF) will contain the world's most powerful laser. The planned experiments to achieve ignition and high gain for the first time in a laboratory will have far-reaching implications for the future of national security, fusion energy, and a host of scientific and technological fields.

The NIF will focus 192 extremely powerful laser beams onto a pea-sized capsule of deuterium and tritium. Two kinds of targets are under study, both of which can be used in the NIF: *direct-drive targets*, in which a spherical capsule containing deuterium and tritium is struck directly by the laser beams; and *indirect-drive targets*, using a capsule inside a small, thin-walled cylindrical container made from high-atomic-number materials such as gold or lead. The container, called a *hohlraum*, converts the driver beams to x-rays which compress the fuel capsule.

The NIF will also be able to achieve more dramatic results in high-energy-density physics than have been possible with lasers to date. The key physics of certain types of stars, of supernovas, of fluid

dynamics and the interior of a nuclear explosion can be revealed, allowing confirmation of computer models and answering questions of nuclear stockpile reliability, fusion energy, and astrophysics.

At a final cost of \$2.0 billion, the football stadium-sized facility would be the centerpiece of the nation's inertial confinement fusion research community, leading a worldwide effort to understand the challenging world of high-energy-density physics and possible fusion-energy production. But leading the world into a nuclear testing-free world is also a vital element of the NIF. Its key role in what is described as *science-based stockpile stewardship* responds to a search for alternative means of maintaining confidence in our nuclear deterrent without nuclear testing. The NIF and other elements of the stewardship program allow researchers the insight to help maintain the stockpile in a safe and reliable fashion.

Answers to important energy questions will be possible with the NIF, as one of the key challenges in inertial fusion research is to achieve ignition of the deuterium-tritium fuel capsule, and further, to show a modest gain in the energy produced. Once researchers can show that they produce more energy than the lasers used to create ignition, the inertial confinement community will have vital information on which to base the next step in its fusion energy development.

Groundbreaking research has been under way at Lawrence Livermore National Laboratory using the Nova laser and at the University of Rochester with the Omega upgrade laser, but the boost in technology achieved by NIF (40 times more energy and 10 times the power of nova) will offer information on inertial fusion ignition and gain that is critical to the next era of ICF progress. It will determine the minimum drive energy (and cost) necessary to ignite an inertial fusion target, thus allowing evaluation of the viability of inertial fusion as an energy source. Since the NIF will be able to reproduce conditions that exist in stellar interiors, it will become an important new tool for laboratory astrophysics. It will also allow important experiments in other fundamental sciences.

Initial operation of the first four of 192 beams began in April 2003.

NIF Design Requirements. The design for NIF takes into account the requirements and requests of three main user communities. The top-level performance requirements for the NIF were driven by the indirect- (x-ray-) drive, ICF mission. Those requirements are as follows:

- 1.8 MJ of laser pulse energy on target
- Flexible pulse shaping (dynamic range >50)
- Peak power of 500 TW
- Pulse wavelength in the ultraviolet (0.35 μm)
- Beam power balance better than 8% over 2 ns
- Pointing accuracy $< 50 \mu\text{m}$
- Compatibility with cryogenic and noncryogenic targets
- Ability to do 50 shots per year, each with a yield of 20 MJ, for a total 1200 MJ annual yield
- Maximum credible DT fusion yield of 45 MJ
- Ability to perform classified and unclassified experiments

In addition to these capabilities, weapons physics users desire the highest possible peak power for short pulses ($>750 \text{ TW}$ at 3ω for 1 ns) in order to reach high temperatures and a range of pulse lengths from 0.1 to 20 ns for a wide variety of experiments. These users want bright sources for experiments requiring x-ray backlighting, with small spots at high temperatures (half the energy in a 100- μm spot, and about 95% of the energy at 200 μm). The beams for these backlighters must be pointed a few centimeters away from the center of the main target chamber.

Weapons effects users want the ability to locate arrays of laser targets several tens of centimeters from the target chamber center, as well as 1ω and 2ω capability. Their other requirements include access to the chamber for large, heavy test objects, a well-shielded diagnostics area for testing electronic systems, and no residual light on the test objects.

The NIF target chamber will also have ports that allow the beams to be placed in the proper location for direct-drive ICF experiments and for tetrahedral hohlraums as well as for the baseline cylindrical

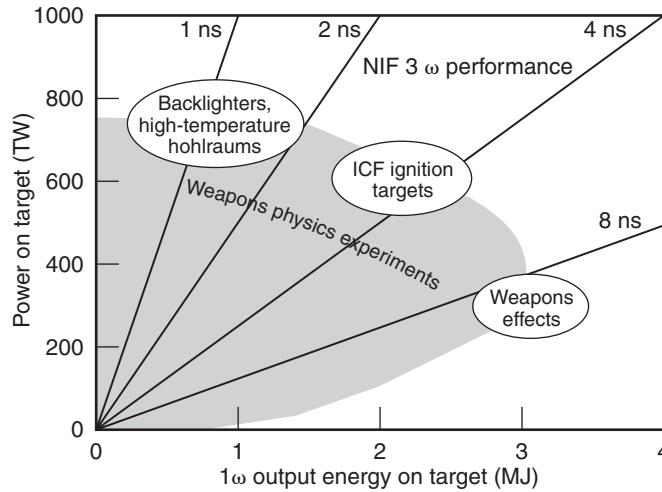


FIGURE 5-56 National Ignition Facility (NIF) users have identified important experiments spanning over a wide range of operating conditions.

hohlraums. All these requirements mean that the NIF must accommodate experiments spanning a wide range of operating conditions (see Fig. 5-56).

NIF Description. The NIF laser system provides routine operation at 1.8 MJ/500 TW in an ignition-target-shaped pulse and has a wide range of operation to meet other user requirements. The laser uses neodymium glass amplifier slabs, with 192 beams in a multipass architecture. Frequency conversion is to the third harmonic: 3ω (350 nm). The laser has adaptive optics (deformable mirrors) to control the beam quality and uses kinoforms and smoothing by spectral dispersion (SSD) to control the beam quality on the target.

Figure 5-57 shows the laser and target area building, as they appear in the Title I Design (submitted to DOE). The overall floor plan is U-shaped, with laser bays forming the legs of the U, and switchyards and the target area forming the connection. The NIF will contain 192 independent laser beams or “beamlets” measuring 40×40 cm each. Beamlets are grouped into 2×2 “quads,” which are stacked two high in 4×2 “bundles” (thus eight beamlets per bundle). These bundles are grouped six each into four large “clusters,” two in each laser bay, for a total of 192 beamlets (8 beamlets per bundle $\times$ 6 bundles per array $\times$ 4 clusters). The 192 laser beamlines will require more than 7000 discrete, large optical components (larger than 40×40 cm) and several thousand smaller optics. Beams from the laser will strike a series of mirrors, which will redirect them to the large target chamber shown on the right side of Fig. 5-57. The building for the NIF will be about 100 m wide (122 m including the capacitor bays), and about 170 m long.

In deciding how many beams for the NIF, two conditions have to be met. First, there has to be enough beam area facing the target to deliver the required energy. The maximum safe 3ω fluence for an ignition target pulse is about 9 J/cm^2 . The maximum practical single beam area is about 1300 cm^2 , which would deliver about 11 kJ/beam on the target. At 11 kJ per beam, at least 164 beams are required to put 1.8 MJ on target. Second, the conditions required by indirect- (x-ray-) drive targets are essential factors. These targets (cylindrically symmetric hohlraums) require twice as many beams in the outer cone as in the inner cone, illumination from two directions, and eight or more beam spots per cone. When we multiply these factors together, we find that the beam count must be divisible by 48. The smallest system that satisfies these two conditions is $4 \times 48 = 192$ beams.

It turns out that it is also very convenient to transport these beams in 48 (2×2) clusters, and that this configuration is also compatible with direct-drive uniformity requirements. Finally, 192 beams at 9 J/cm^2 provides 2.2 MJ, a full 20% margin for baseline operation of 1.8 MJ.

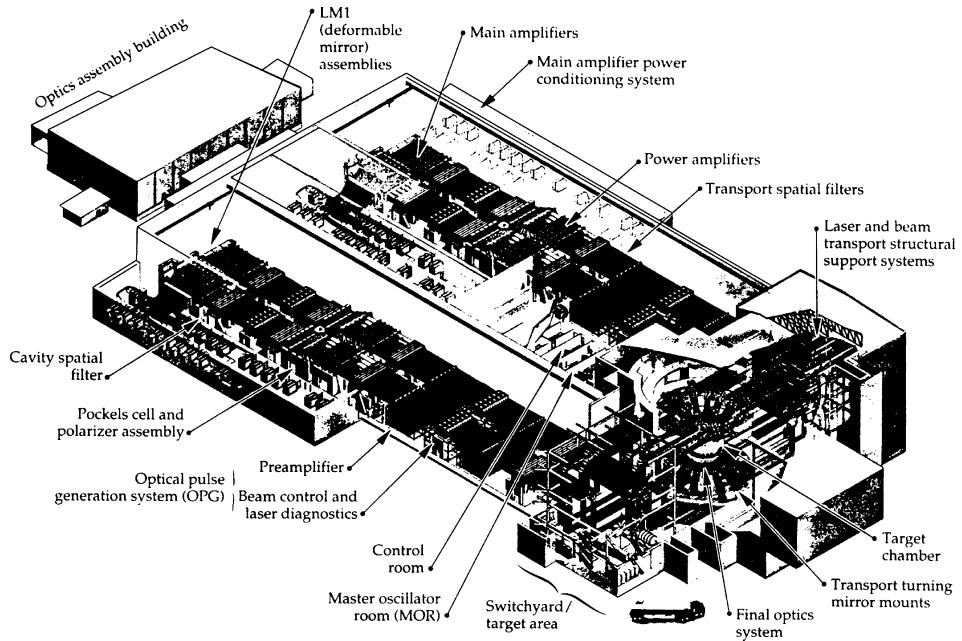


FIGURE 5-57 Layout of the laser and target area building. The two pairs of large beamlet clusters running the length of each laser bay are essentially identical to each other.

Master Oscillator System. The laser pulse is produced in the master oscillator room, where a fiber-optic oscillator generates a weak, single-frequency laser pulse. A phase modulator puts on bandwidth for smoothing by SSD and suppressing stimulated Brillouin scattering. Each pulse is then launched into an optical fiber system that amplifies and splits the pulse into 48 separate fibers. The optical fibers carry the pulses to 48 low-voltage optical modulators very similar to those used in high-bandwidth fiber communication systems. These modulators allow temporally and spectrally shaping of each pulse by computer control. An optical fiber then carries each nanojoule, 1- μm pulse to a preamplifier module (PAM).

Preamplifier Module. Optical fibers carrying the pulses from the master oscillator room spread out to 48 preamplifier modules, located on a space frame beneath the transport spatial filters. Each preamplifier has a regenerative amplifier followed by a flashlamp-pumped four-pass rod amplifier. The preamplifier is a two-stage system, designed as a self-contained assembly, that can be pulled out and replaced as needed. The preamplifier brings the pulse to about 10 J, with the spatial intensity profile needed for injection into the main laser cavity. Before entering the main cavity, the output from the preamplifier is split four ways. These four pulses are injected into the four beams that form each of the 48-beam quad arrays.

Main Laser System. Figure 5-58 shows the layout of the main laser components of a NIF beamlet. These components take a laser pulse from the preamplifier to the final optics assembly. A laser pulse from the preamplifier enters the main laser cavity when it reflects from a small mirror labeled LM0. This mirror is located near the focal plane of the transport spatial filter. The 40-cm-diameter pulse exits the transport spatial filter, traveling to the left, and passes through the power amplifier, containing five amplifier slabs.

The beam then enters the periscope assembly, which contains two mirrors (LM2 and LM3) and a switch (a Pockels cell and a polarizer). The pulse reflects off LM3 and the polarizer before passing

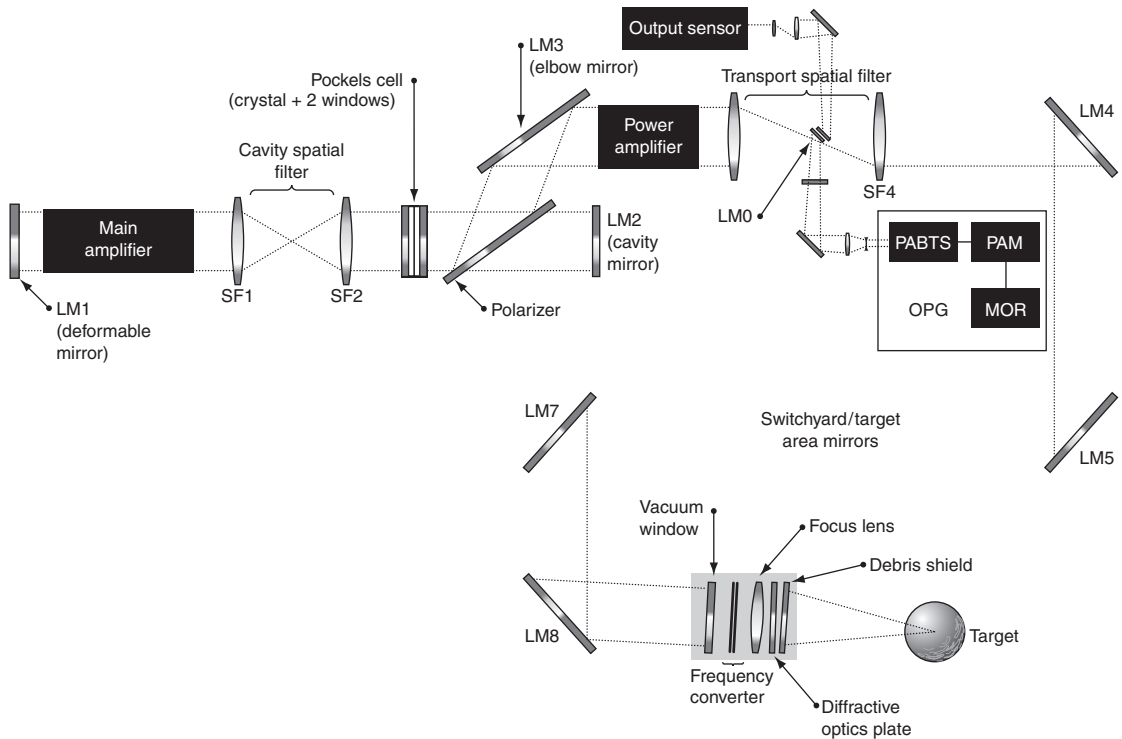


FIGURE 5-58 Schematic diagram of the NIF laser system.

through the Pockels cell. It goes through the cavity spatial filter and the 11-slab main amplifier, and then reflects from a deformable mirror with 39 actuators. After once again passing through the main amplifier, the beam comes back through the cavity spatial filter to the periscope assembly. Meanwhile, the Pockels cell has been fired to rotate the polarization, so the beam passes through the polarizer and strikes mirror LM2, which redirects it back through the cavity spatial filter for another double pass through the main amplifier. The beam returns to the periscope assembly, passes through the deenergized Pockels cell, reflects off the polarizer and LM3, and is further amplified by the power amplifier. Now the pulse passes through the transport spatial filter on a path slightly displaced from the input path. The output pulse just misses the injection mirror LM0 and enters the switchyard and beam transport area.

Switchyard and Beam Transport. Between the switchyards and the target chamber room, the beams are in two 2×2 arrays: the 4×2 bundles are split into 2×2 quads, moving up or down to one of the eight levels of the switchyard. Each pulse now travels through a long beam path, reflecting off of several transport mirrors before reaching the target chamber. The transport mirrors can be moved to create the beam configuration needed for direct- or indirect-drive experiments. For indirect drive, mirrors send the beams straight up or straight down to make the cones coming into the top and the bottom of the target chamber cylinder. For direct drive, 24 beams are directed to circumferential positions around the target chamber by moving two mirrors in each of these beamlines. Once the beams reach the target chamber, they enter the final optics assembly.

Final Optics Assembly. The final optics assemblies are mounted on the target chamber. Each assembly includes a vacuum window at 1ω , a cell that includes a frequency converter (two plates of

potassium dihydrogen phosphate crystal) to convert the pulse to 3ω , and the final focusing lens. The cell tips and tilts to tune the frequency converter, and translates along the beam direction to focus the beam on the target. A debris-shield cassette includes the capability of diffractive optics for spot shaping. Once a pulse travels through this assembly, it proceeds to the target in the target chamber.

Other Laser Facilities. Other powerful lasers for ICF experiments include GEKKO XII at Osaka University in Japan, Omega at the University of Rochester in New York, and Nike at the Naval Research Laboratory in Washington, D.C. The GEKKO XII laser used ND: glass laser at $0.53\text{ }\mu\text{m}$, and 12 beams with random-phase-plates implemented. The laser energy is 8 to 10 kJ with a nominally 1.7-ns (full width at half-maximum, [FWHM]) flat-top pulse with a rise time equivalent to a 1- to 1.3-ns Gaussian pulse (FWHM). Achievements at GEKKO XII include the generation of 10^{13} neutrons per shot, a 10-keV plasma with stagnation-free mode compression, and a density compression up to 600 times solid density using a plastic shell target implosion. An upgrade of the GEKKO system to double the output in blue laser light has been implemented. Further, it is envisioned that a solid-state laser could be ultimately developed for use in a power plant using laser diode (LD) pumping of the laser and advanced active cooling techniques. In addition to technical issues involved in this approach, the projected cost for the LD unit must be substantially reduced to achieve economic competitiveness.

LD-pumped solid-state lasers have advanced to driver candidate status due to the recent advances of AlGaAs LD arrays with power of $\sim 4\text{ kW/cm}^2$ (200- μs pulse duration), an efficiency of $\sim 60\%$, and a lifetime of 10^{11} shots. A conceptual power plant design has been carried out for an LD system with 10-MJ blue output, 10% efficiency, and 10-Hz repetition rate. Also, additional constraints incorporated in the design include the following:

To prevent damage, the fluence of the laser beam in the amplifier was restricted to be 32 J/cm^2 at a pulse width of 10 ns.

To prevent parasitic oscillation in the laser disk material, the product of the largest dimension of the disk and the small signal gain coefficient was held ≈ 4 .

Cooling was added to keep the maximum temperature difference and the peak temperature in the disk below 10% and 30% of the transition or melting point of the laser material, respectively.

The maximum thermal stress in the disk was held to $\approx 20\%$ of the fracture strength to ensure optical quality and long-lifetime operation.

OMEGA Laser Facility. The OMEGA laser is housed in the Laboratory for Laser Energetics (LLE) at the University of Rochester, New York. This facility is part of the national laser fusion effort within the U.S. Department of Energy and has as its main mission direct-drive laser fusion research in support of the upcoming National Ignition Facility (NIF) at the Lawrence Livermore National Laboratory (LLNL). The OMEGA laser system has been operating since May 1994 (McCrory 1994, Seka et al. 1997, Sourers et al. 1996). This 60-beam UV laser facility can cover a wide variety of experiments related to ICF. The facility also supports indirect-drive laser fusion research under the direction of LLNL and the Los Alamos National Laboratory (LANL).

The OMEGA laser system (Boehly et al. 1992) is a Nd:glass laser system with frequency conversion to the third harmonic with a UV (351-nm) energy capability on target in excess of 30 kJ for pulse durations of $\leq 2\text{ ns}$. The system's 60 beams (30-cm beam diameter) are arrayed symmetrically around the 3.3-m-diameter target chamber and are focused onto the target by 60 near-diffraction-limited $f/6$ lenses.

Laser pulses with a wide variety of predetermined, temporal UV output pulse shapes have been produced by a versatile pulse-shaping system. These shapes range from 1- to 3-ns flat-topped pulses to linear ramps (1 to 3 ns), Gaussians (0.2 to 1.2 ns), and a special 2-ns pulse designed for indirect-drive targets.

The demonstrated energy balance of the OMEGA system is $\sim 2\%$ rms for the IR part of the system, $\sim 3\%$ rms in the UV after the conversion crystals, and $\sim 4\%$ to 5% rms on target after the UV transport optics (two turning mirrors, a focusing lens, and a blast shield per beamline). The system's

reproducibility is $<1\%$ rms excluding clearly identifiable flashlamp or PFN malfunctions. The measurement accuracy of the beam energies (including residual fundamental and second-harmonic output from the conversion crystals) is $\sim 0.8\%$ rms.

The OMEGA laser system has three complete driver lines with corresponding pulse-shaping systems. Any one or any combination of two of these driver lines can be used to drive the laser system. The laser system supports the propagation of two different pulse shapes: one for the main experiment (driver) and the other for a backlighter. In this case, one driver line feeds 40 beamlines, while the other feeds the remaining 20. The timing between these two groups of beams can be chosen freely.

Beam smoothing, an essential component for the success of direct-drive laser fusion, is implemented on OMEGA using smoothing by SSD pioneered at LLE. LLE's approach combines continuous-phase-plate technology with polarization rotators and two FM modulators with corresponding spectral dispersion in two dimensions (2-D SSD). Present capabilities are limited to 2- to 3-Å total bandwidth for the two modulators (3 and 3.3 GHz modulation frequency). In addition, a limited number of phase plates and polarization rotators are available primarily for planar target experiments. A full complement of phase plates is available for spherical irradiation experiments, and complete complement of polarization rotators is expected soon. At that time, increased bandwidth (~ 10 Å) and higher FM modulation frequencies (~ 10 GHz) will also be available, allowing an on-target frequency bandwidth of ~ 1 THz in the UV spectrum.

The pointing capability of the OMEGA laser system has been shown to be ~ 10 μm rms on target. This performance has been repeatedly verified with x-ray imaging using pinhole cameras and x-ray microscopes. The stability of the system is sufficient to maintain the above pointing accuracy without adjustments for at least a period of 1 day. Furthermore, beams can be placed in basically arbitrary locations within ~ 1 cm from the target chamber center with essentially the same accuracy (~ 20 mm without corrections; better accuracy is achievable with one iterative fine correction). This capability has been demonstrated on recent indirect-drive hohlraums of cylindrical (two laser entrance holes) and spherical shape ("tetrahedral" hohlraums with four laser entrance holes) with or without additional backlighter targets.

The present OMEGA experiments address irradiation uniformity, implosion physics, hydrodynamic instabilities, laser imprinting, and laser-plasma-interaction physics. In addition, there are experiments carried out by LLNL and LANL relating to indirect-drive laser fusion.

Liquid or solid layers of DT are required to minimize the drive requirements for high-performance, direct- and indirect-drive targets. In a collaboration involving LLNL, LANL, General Atomics (GA), and LLE, the technology required to conduct cryogenic-fuel-layer ICF experiments is being developed at the OMEGA facility (Sourers et al. 1996).

As part of the national effort on cryogenic target development, GA is currently designing a cryogenic target delivery system for OMEGA experiments. The system is shown schematically in Fig. 5-59.

Using this system, polymer shells will be filled with DT up to 1500-atm-pressure, room-temperature equivalent. The shells will be transferred to the vicinity of the OMEGA target chamber in a cold-transfer cryostat. Fuel layers up to 100 μm thick will be formed inside the 1-mm-diameter plastic shells. The capsules will then be inserted into the OMEGA chamber and irradiated by the 60-beam system. Liquid layering is expected to occur inside the chamber with in situ characterization just before the target is shot.

A \$50 million upgrade to the OMEGA laser is planned for completion in 2007. It will add two high-energy Petawatt lasers for advanced fast-ignition experiments.

The NIKE KrF Laser. In order to achieve an attractive laser-driver for inertial fusion energy (IFE), a higher laser efficiency is needed than achieved with current flash-lamp-pumped Nd-glass lasers. One approach that combines a reasonably high laser efficiency (around 5%) and short wavelength (UV) is the KrF laser (Sethian et al. 1997).

NIKE is a multikilojoule krypton-fluoride (KrF) laser that has been built at the Naval Research Laboratory (NRL), Washington, D.C. to study the physics of direct-drive inertial confinement fusion. Both the two final amplifiers of the NIKE laser are electron-beam-pumped systems. This design places strong reliance on the pulsed electron beams to drive the amplifiers. The smaller of the two has a 20×20 -cm aperture and produces an output laser-beam energy in excess of 100 J (Fig. 5-60).

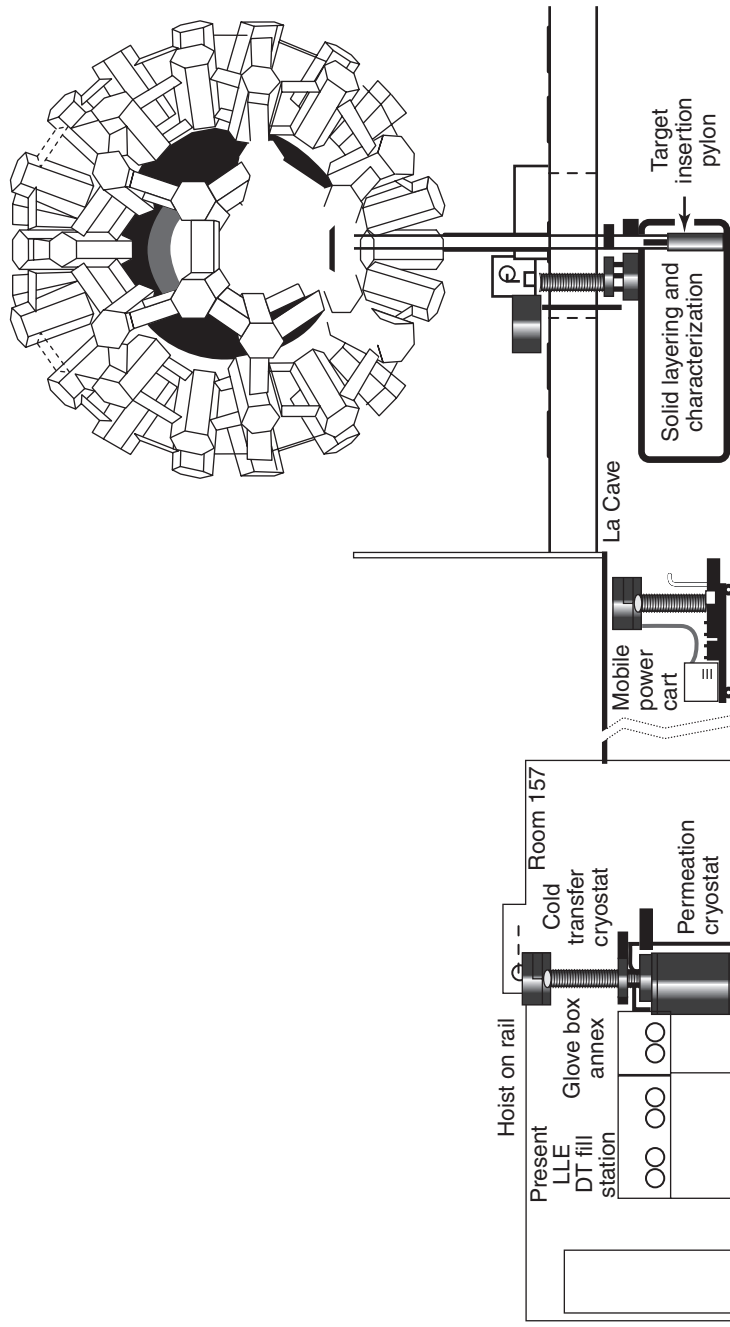


FIGURE 5-59 Schematic diagram of OMEGA cryogenic target system.

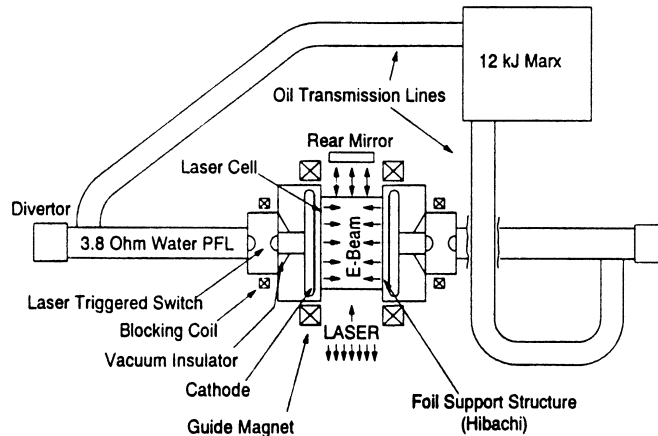


FIGURE 5-60 The NIKE 20-cm amplifier. The paths of the laser beam and electron beams are shown in this figure. The laser beam enters the laser cell (from the bottom of the figure), is amplified once, is reflected off the rear mirror, and is amplified again as it passes back through the cell. The 60-cm amplifier uses a similar concept.

This 20-cm amplifier uses a single 12-kJ Marx generator to inject two 300-kV, 75-kA, 140-ns flat-top electron beams into opposite sides of the laser cell. The larger amplifier in NIKE has a 60×60 -cm aperture, and amplifies the laser beam up to 5 kJ. This 60-cm amplifier has two independent electron-beam systems. Each system has a 170-kJ Marx generator that produces a 670-kV, 540-kA, 240-ns flat-top electron beam. Both amplifiers are complete, are fully integrated into the laser, meet the NIKE system requirements, and are used routinely for laser-target experiments. To obtain sufficiently high target gains for a direct-drive fusion reactor, a KrF laser based on the NIKE concept must take advantage of three optimizations. First, the laser-beam illumination on the pellet has to be extremely uniform. High-mode beam nonuniformities in the range of 0.2% rms are required, along with low-mode nonuniformities of about 1%. These nonuniformity levels have already been achieved with NRL's KrF laser. Second, the rocket efficiency has to be maximized by depositing the laser energy deeply into the pellet. KrF, with $0.25\text{-}\mu\text{m}$ light, deposits at about twice the plasma density as a solid-state laser at $0.34\text{ }\mu\text{m}$. Third, the target gain is optimized by "zooming" the laser beam inward during the implosion, thereby better matching the laser spot size to the pellet diameter. This optical zooming is achieved in a straightforward fashion with KrF lasers. There are also several engineering challenges in developing a laser of this type with sufficient energy, repetition rate, reliability, and economy for a practical reactor. Some of these challenges are the lifetime of the emitter and pressure foil in the electron-beam pumped amplifiers, the ability to clear the laser gas between pulses without sacrificing beam quality, and the overall efficiency of the system.

Accelerator and X-Ray Fusion. In addition to lasers, accelerators and x-rays are considered strong candidates for ICF drivers, especially for future reactors. We will consider both briefly here.

Light-Ion Accelerators. Pulsed-power-based light-ion particle accelerators (Bluhm et al. 1992, Mehlhorn 1997, Quintenz et al. 1996) have been under development as ICF drivers since 1972. The technology produces a beam that couples well to matter and is scalable to very high power levels. Light-ion ICF uses pulsed-power accelerators that compress electrical energy in both space and time, and then convert the energy into directed ion beams in a device called an *ion diode*. The ion power and focusability can be improved by using two acceleration gaps, in a device called a *two-stage diode*. Self-pinch transport is an attractive option for propagating the beam from the diode to the target. A stand-off distance of several meters is required to protect the diode from the fusion target blast. Finally, the

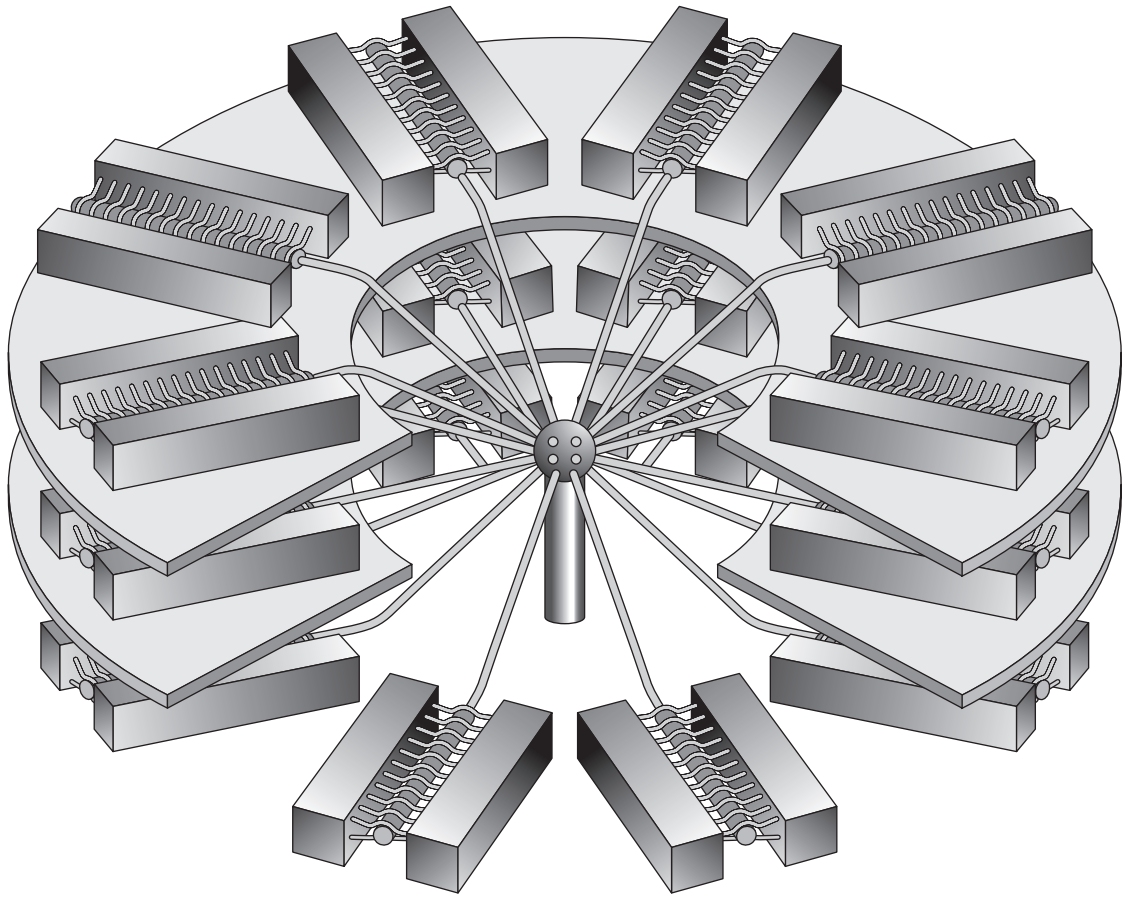


FIGURE 5-61 Z-facility showing modules arranged in three tiers.

light-ion driver beam is focused onto the ICF target, producing x-rays that drive the capsule to fusion conditions, resulting in a large fusion energy yield. Light-ion research is focused on developing the basic physics and technology scalings that could enable the construction of a high-yield facility (HYF) or engineering test facility (ETF) for IFE as shown schematically in Fig. 5-61. A light-ion IFE facility would comprise many modular pulsed-power accelerators that are compact, efficient, and low cost. Ion-beam divergence is the ratio of the transverse to longitudinal beam velocity, and is a measure of the focusability of a beam. Each module would have a two-stage ion diode comprising an injector section that accelerates lithium ions to ≤ 24 mrad at 9 MeV, a second stage of acceleration to reduce the divergence to ≤ 12 mrad at 35 MeV (by adding only longitudinal momentum), and a self-pinch transport channel that can propagate a 12-mrad lithium beam up to 4 m to the target. The ion beams from these modules would be focused onto a single target to supply the total power and energy requirements for ICF with sufficient uniformity to drive a symmetric implosion. Considerable progress in this technology has been made on accelerators at Sandia National Laboratory (SNL) in New Mexico and the Research Center Karlsruhe (FZK) in Karlsruhe, Germany. Proton beams have been focused to peak intensities of 1 TW/cm^2 at FZK and 5 TW/cm^2 at SNL, while SNL has focused a lithium beam to $\sim 2 \text{ TW/cm}^2$. Focused lithium beams have heated a target to almost 70 eV at specific energy densities that are comparable to the initial pulse level of an HYF target. At present, the key technical challenges

for the light-ion program are (1) developing low-emittance ion sources that can provide 1–2 kA/cm², (2) reducing total ion beam emittance, (3) increasing total ion beam brightness by two-stage acceleration, and (4) developing a simple robust ion beam transport system. The modularity of the light-ion approach provides a cost-effective development path for an HYF/ETF. The key technical issues can be studied on existing smaller accelerators such as SABRE at SNL and KALIF at FZK; the scaling of the results can be tested on existing larger accelerators such as HERMES-III at SNL, leading to the construction of a single HYF module and the eventual construction of a full HYF/ETF. This technology is now used to produce x-rays for inertial confinement fusion pellet implosions.

Z-Pinch X-Ray Fusion. Since October 1996, the Z-facility (previously known as PBFA-II) at Sandia National Laboratories (Albuquerque) has operated in a Z-pinch configuration to provide experimental data to the Stockpile Stewardship Program of the National Nuclear Security Administration (NNSA). In the new configuration, electrical energy is coupled into the kinetic energy of a magnetically-driven imploding plasma, formed by passing a large current through an annular array of wires. An intense, short pulse of soft x-rays is created when the imploding plasma stagnates on axis. These x-rays are then directed onto the surface of a fusion capsule, heating and compressing it. In a fusion power plant based on Z-pinch x-rays, an array of accelerators would deliver x-rays sequentially to a target chamber at about 0.1 Hz.

Work is proceeding on the design and testing of key systems for the refurbishment of the Z-facility (to be known as ZR) to be complete in 2006.

Heavy-Ion Accelerators for ICF. High-energy particle accelerators are considered as potential ignition devices for inertially driven fusion targets. Relativistic particle beams have the advantage that intrabeam forces are reduced by the cancellation of electric repulsion by magnetic attraction between two particles moving in quasiparallel paths near each other. Also, with large beam energies, huge peak powers can be achieved with relatively small currents. For instance, at 10 GeV, a peak power of 400 TW corresponds to a peak current of 40,000 A (or the combined effect of 40 beams each with 1000-A current), entirely within the realm of achievable currents for a variety of particles.

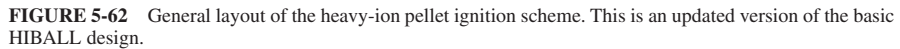
A key difficulty in the use of high-energy beams is the requirement of depositing most of the beam energy within the relatively thin outer shell of the target. To meet this goal, one can use heavy ions, which even at several GeV have a range smaller than ~ 1 g/cm² or, alternatively, use relativistic electrons, but with an intermediate, “energy-efficient” conversion into soft-x-ray photons (~ 0.1 to 1.0 keV). The x-rays then drive the target implosion (so-called indirect drive).

The key figure of merit in colliding-beam devices is *luminosity*. This requires high current densities and, in most designs, a large number of high-energy particles packed in a tight bunch. These requirements are not unrealistic, however. For instance, the high current reached many years ago in the CERN-ISR has spurred these ideas on. The energy stored in each beam of the ISR has now reached 5×10^6 J using a beam of 50 A at 31.5 GeV for the duration of 3.3 μ s, roughly the correct order of magnitude of energy for ICF ignition. There are, however, two important differences when comparing ICF requirements with CERN-ISR: first, the beam duration for ICF should be $\leq$ ns in order to achieve the instantaneous power required and second, protons must be replaced by heavier ions (e.g., lithium) in order to ensure a sufficiently short range. These extrapolations are by no means trivial.

To operate a large-scale ICF power station, beams with a total energy delivered to the target of several 10^7 J are required. Since the beam duration must follow the implosion time, the length of the pulse must be a few nanoseconds. Hence, the peak power will reach $\sim 10^{15}$ to 10^{16} W, or several hundred TW. More detailed calculations with 10-GeV bismuth ions focused to a 2-mm radius target, show that an energy ~ 5 MJ and a peak power of 300 TW are appropriate for a gain $G \sim 80$, sufficient for a commercial-grade reactor.

Another difference between current accelerator experiments and ICF involves the environment in the vicinity of the final focus in ICF. While peak currents in excess of 10^3 A have been focused to submillimetric spots in colliders at hard vacuum without problems, the reactor vessel will have vacuum conditions of 1 to 10 torr. Also, the large amount of radiation emitted by the imploding pellet could “backfire” on the incoming beam.

We can now give a short description of the whole scheme (see Fig. 5-62). The initial acceleration of the singly ionized heavy ions uses a linac structure delivering about 50 mA of current during a



The U.S. Heavy Ion Fusion Program currently places great emphasis on cost reduction since high capital cost appears to be one of the disadvantages of this approach. The search for lower cost encompasses two main activities: (1) the exploration of new directions in beam physics, plasma physics, and target design (mostly scientific issues), and (2) development of improved materials, better hardware, and better fabrication techniques (mostly technological issues). As the understanding of the science of heavy-ion fusion continues to improve, several developments aid in cost reduction. One involves beam propagation in the target chamber. Recent numerical studies show that beam neutralization techniques will be effective. Beam neutralization allows a reduction in ion kinetic energy, leading to a significant reduction in accelerator cost. Target physics, including fast ignition, appears to be another scientific area where cost breakthroughs are likely. Important cost reductions in most areas of technology also appear feasible. A heavy-ion induction accelerator consists of a number of components: ion sources, an injector, ferromagnetic cores, insulators, magnetic and electric focusing lenses, pulsed, and diagnostics and controls. Although the ion sources by themselves are not costly, improved sources can lead to lower cost in the remainder of the accelerators; consequently, several new source technologies, including aluminosilicate, contact ionization, gas, laser, and spark sources, are being explored. The cost of ferromagnetic materials for the induction cores ranges from a few dollars per kilogram to more than \$100/kg. Techniques are under study to insulate and anneal inexpensive materials to make them suitable for accelerator applications. Traditionally, insulators for accelerators are made from alumina.

The joints to metal are made by brazing. Large alumina insulators needed for fusion accelerators are costly, as is brazing. Thus, insulators that can be cast with metal attachments are being developed. Innovative superconducting magnetic lenses with an emphasis on designs that can be inexpensively fabricated are being developed. Conventional and advanced pulsers for heavy-ion fusion are also under study. In terms of conventional pulsers, most work is with thyatrons, exploring methods to optimize them for our application. Advanced pulsers employ solid-state devices and magnetic pulse compression. Finally, improved diagnostics and control systems are being developed. Improved ability to sense and control the beam leads to smaller less expensive accelerators.

Fusion Breeder. The physical basis of the fusion breeder (also called the fusion-fission hybrid) is the prolific production of high-energy neutrons in fusion reactions. These neutrons can be used to breed fertile fuel (plutonium and ^{233}U) for use in conventional fission reactors. Each D-T fusion reaction produces 14 MeV of total energy, or approximately 4 times as many neutrons per unit of energy as a fission event (~ 3 neutrons per 200 MeV). The 14-MeV fusion neutrons produce additional neutrons in the breeder blanket by neutron-multiplying nuclear reactions, by fast fissions in fertile materials, and/or by nonfissioning reactions such as $(n, 2n)$ reactions in beryllium.

5.4.10 Breeder Types

Two different fusion breeder types based on these two methods of neutron multiplication have emerged. One uses a fast-fission blanket and the other uses a suppressed-fission blanket. In the former, the D-T fusion source is surrounded by a blanket of fertile material (^{238}U and/or ^{232}Th) and a lithium compound for tritium breeding as shown in Fig. 5-63. Fast fissions in the fertile material multiply both the fusion energy (3 to 10 times) and the neutrons (approximately two to four neutrons per fusion neutron). One of the neutrons is needed to breed tritium from lithium and the remainder are available for breeding fissile fuel.

In the suppressed-fission blanket (see Fig. 5-63), a nonfissioning, neutron-multiplying material such as beryllium replaces most of the fertile fuel in the blanket. Consequently, the fast-fission source of neutron multiplication is replaced with a nonfissioning source of neutrons. To further suppress the fissioning of the bred ^{233}U , its concentration is limited to about 1% or less of the fertile fuel. Thus, the breeding blanket must be designed to allow for online, or quick, low-cost removal and recovery of the bred fissile fuel. The low fission rate results in superior overall reactor safety characteristics. Also, a much lower fission product inventory and a lower decay afterheat results.

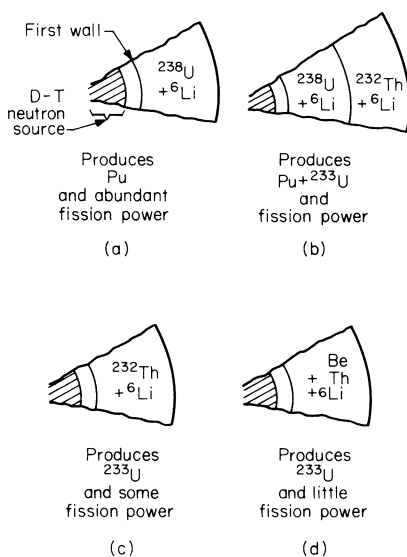


FIGURE 5-63 Methods of neutron multiplication: (a) ^{238}U fast-fission blanket; (b) $^{238}\text{U} + ^{232}\text{Th}$ fast-fission blanket; (c) ^{232}Th fast-fission blanket; (d) ^9Be fission-suppressed blanket.

Fuel Cycles. The suppressed-fission blanket can produce plutonium rather than ^{233}U by substituting uranium carbide for the thorium, but then the number of client LWRs that can be fueled is lower. For example, for a recent design with a 2100-MW fusion driver (4000 MW_{nuclear}), the suppressed fission design could support 14 LWRs or 4000 MW_{nuclear} each when producing ^{233}U but only about 9 when producing plutonium. The advantages of higher LWR support motivate the desire for new fuel-reprocessing technologies for ^{233}U and thorium. Two technologies being considered are the conventional, aqueous THOREX reprocessing technology and a pyrochemical process that uses magnesium dissolution of the thorium leaving the ^{233}U as a precipitate.

Breeder Role. Fusion breeders are best understood in the context of a symbiotic fusion-breeder-fission-burner system that generates electricity. The fusion breeder

would be incorporated into a fuel-cycle complex along with fuel-reprocessing plants, fuel-fabrication facilities, and possibly a waste-disposal facility—all in a safeguarded area. The fissile fuel produced in the fusion breeder is recovered by reprocessing, mixed with fertile fuel, fabricated into fuel rods, and shipped to the fission burner reactors. The spent fuel from the burner reactors is returned to the safeguarded fuel-cycle complex where the remaining fissile fuel is separated from the radioactive waste material.

Support Ratio. The thermal support ratio is defined as the nuclear (or thermal) power of the client fission reactor (e.g., LWR) divided by the nuclear power of the fusion breeder. A high thermal support ratio is advantageous because such a breeder could fuel a large number of fission reactors, thus having a large commercial impact. Also, for large support ratios, only a small fraction of the symbiotic system's cost and electricity generation is attributed to the addition of a fusion breeder into the existing electricity generation system (typically 15% of the overall capital cost and about 5% of the overall electricity generation). Because relatively little power would be produced within the safeguarded fuel-cycle park itself, the breeders could be in a remote location.

Thermal support ratios for fusion breeders range from 4 to 45, depending on the choice of fusion blanket and client thermal converter (e.g., LWRs or advanced converter reactors). The following support ratio estimates are typical: uranium fast-fission blankets produce enough plutonium to support 4 to 6 LWRs; thorium fast-fission blankets produce enough ^{233}U to support 8 to 12 LWRs, or 14 to 28 advanced converters; uranium suppressed-fission blankets produce enough plutonium to support 9 to 11 LWRs; thorium suppressed-fission blankets produce enough ^{233}U to support 12 to 16 LWRs, or 35 to 45 advanced converters.

The variations in these support-ratio estimates depend on the specifics of the fusion-blanket designs, the type of client fission reactor, and fuel-cycle choices.

These support ratios can be put in perspective by comparing them to the support ratios of a liquid-metal fast-breeder reactor (LMFBR). A typical LMFBR does not produce enough excess fissile fuel to support even one LWR of equivalent nuclear power. Furthermore, the LMFBR must also produce fissile fuel to satisfy the fissile inventory requirement of additional LMFBRs. Consequently, LWR support is not an effective mode of LMFBR operation. The fusion breeder, on the other hand, requires no initial fissile inventory, and its tritium inventory is quite low.

Performance Requirements. Fusion breeders can operate economically with significantly lower fusion performance and higher cost than fusion electric power plants. We consider two performance indicators for magnetic fusion devices: plasma energy gain Q and first-wall fusion neutron wall loading Γ . Q is defined as the ratio of the fusion energy produced to the input energy required to heat and sustain the plasma (supplied by relatively expensive beams of energetic neutral atoms and/or rf heating systems). The fusion neutron first-wall loading in megawatts per square meter is indicative of the blanket power density.

Several conceptual design studies of fusion reactors have shown that pure fusion power plants will require $Q > 20$ and $\Gamma > 3 \text{ MW/m}^2$ to produce competitive electricity. Although high Q is preferred, fusion breeders will only require Q values from 2 to 6 and $\Gamma > 1 \text{ MW/m}^2$ to economically produce fuel for LWRs. Fusion breeders with fast-fission blankets can operate in the lower- Q regime (i.e., Q from 2 to 4); with suppressed-fission blankets, Q values above 6 are required.

These performance requirements also apply to ICF systems. For example, ICF hybrid studies have shown that a driver-efficiency target gain product ($\eta_D G$) of about 6 is acceptable while an $\eta_D G$ product greater than 20 is required for fusion electric power generation. These lower performance requirements can allow relaxation of technology requirements.

5.4.11 Progress toward Attainment of Controlled Fusion

Figure 5-64 illustrates the progress of magnetic fusion toward the goal of controlled nuclear fusion. In this figure, the kinetic temperature of a pure deuterium-tritium plasma (ions and electrons taken to have the same temperature) is plotted against the density-confinement time product $n\tau$. In these coordinates, ignition occurs at any point on or above the locus (top of the diagram) at which the thermonuclear energy gain exceeds 30.

Loci of constant gain (fusion power density at 17.6 MeV per fusion, divided by the external power density required to maintain the plasma temperature against losses) are plotted over a wide range (from

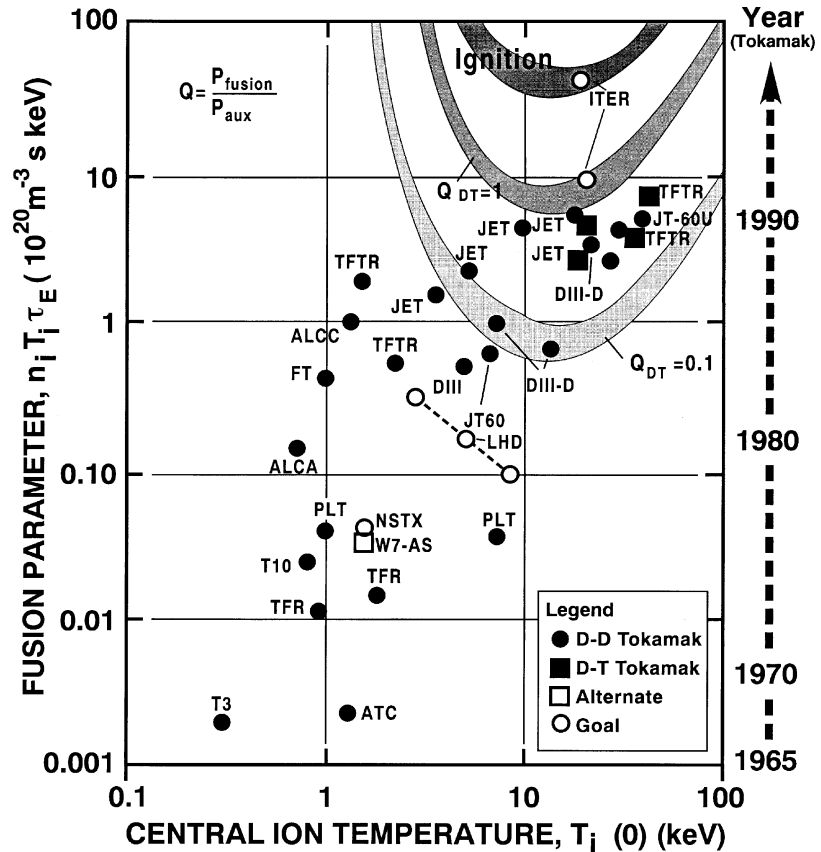


FIGURE 5-64 Progress in magnetic fusion. [Legend: ALCATOR (ALCA and ALCC) = high-field tokamak, MIT; ATC = adiabatic toroidal compressor tokamak, Princeton; DIII = doublet tokamak, GA Technologies, Inc.; JET = Joint European Torus, Culham, U.K.; NSTX = National Spherical Tokamak Experiment, Princeton; START = Spherical Tokamak, Culham, U.K., Lawrence Livermore Laboratory; ORMAK = Oak Ridge National Laboratory tokamak; PLT = Princeton Plasma Physics Laboratory, large-torus tokamak; TFR = tokamak Fontenay-aux-Roses, France; TFTR = tokamak fusion test reactor, Princeton; ITER = international thermonuclear experimental reactor (proposed international project—status is uncertain due to global financial issues); 22 = IIB-mirror reactor, Lawrence Livermore National Laboratory.]

10^{-8} to 10) over which fusion confinement experiments have operated, or are projected to operate. The values of the operating areas of specific experiments are identified by the abbreviated symbols for several confinement devices. The highest attained kinetic temperature achieved is in mirror and certain pinch experiments where about 30 keV, equivalent to 300 million degrees absolute, has been achieved.

Unique Resource Requirements. Table 5-19 lists a number of materials needed to support a nuclear-fusion power economy capable of delivering a total power of a million megawatts (third column) compared with the forecast U.S. demand for these materials in the year 2000 and estimates of world reserves. According to this table, there appears to be no resource limitation that would prevent the extensive development of fusion power. However, the demands for certain materials (beryllium, copper, helium, lithium, vanadium, niobium, and molybdenum) would require additional production capacity, and exploration for new sources would be required for beryllium, lithium, vanadium, niobium, and molybdenum.

TABLE 5-19 Estimated Quantities of the Unique Resource Requirements Associated with Fusion Power

Material	Application	Inventory for 10 ⁶ MW capacity, metric megatons	Forecast U.S. demand for year 2000, metric megatons	World reserves, metric megatons	Quantity contained in upper 10 m of Earth's crust metric megatons	Ratio of quantity in Earth's crust to world reserves
Beryllium	Neutron multiplication and PFCs*	0.046	0.002	0.09	1.5(4) [†]	1.6(5)
Aluminum	Coil conductor	1.0–2.6	33	1000	5.1(8)	5.1(5)
Copper	Coil and firewall conductor	3.2–8.6	7	300	2.3(5)	7.6(2)
Helium	Refrigerant	0.04–1.1	0.02	1	4(3) [‡]	4(3)
Lithium	Fuel, coolant	0.95–1.5	0.014	0.8	2(5) [§]	2.5(5)
Titanium	Structure, SC	0.5	2	150	2.8(7)	1.9(5)
Vanadium	Structure, SC	2.4	0.03	10	6.6(5)	6.6(4)
Niobium	Structure, SC	3.3	0.01	10	9.4(4)	9.4(3)
Molybdenum	Structure	2.8	0.09	5.4	6.2(3)	1.1(3)
Tin	SC	0.3	0.1	7.2	1.4(4)	2(3)
Lead	Shielding	11	2.5	95	9.4(4)	1(3)

*PFCs = Plasma-facing components.

[†]To be read as 1.5×10^4 .[‡]In the atmosphere.[§]There is approximately the same quantity of lithium in seawater.^{||}SC = Superconductive materials.

Many of these materials will require new fabrication technology and the development of measures against structural deterioration under neutron bombardment and other radiation effects.

Safety and Environmental Considerations. Fusion power has long been considered preferable to fission power from the viewpoints of safety and environmental impact. Fusion reactors, in large part because they are concerned with fuels and reaction products of low atomic number, have lower potential for the release of radioactive materials. Also, the fusion process does not inherently result in radioactive products that require long-time storage like the fission-product waste from fission reactors. But more detailed examination of the processes involved in fusion shows that there are several key points at which stringent radiation safety measures must be taken.

The principal environmental hazard of fusion reactors employing the D-T-Li cycle is the release of tritium during normal operation or in the event of an accident. The avenues of escape of tritium during normal operations are the lithium blanket and its associated containment structure, fluid piping, and heat-exchanger tubing. All these elements operate at extremely high temperatures, with corresponding high permeation rates.

Fusion safety standards have been developed that require tritium releases to be maintained below 10 Ci/day in routine operations and considerably higher values are proposed in the event of a severe accident. It is assumed, on the basis of experimental evidence, that essentially all elemental tritium escaping to the environment will become oxidized. Oxides are several thousand times more hazardous than the element itself.

In a 1000-MW fusion plant, the internal throughput of tritium of about 4×10^6 Ci/day would require a containment factor of 99.9999% to limit the escape to 10 Ci/day. While such a factor is feasible, its attainment at acceptable costs must be demonstrated before large-scale fusion plants employing tritium can be demonstrated to be cost effective.

Fusion neutrons will also cause activation of the first-wall and blanket materials. This radioactivity can be minimized by judicious selection of materials. The overall biological-hazard potential of a typical fusion blanket system is lower than that of a liquid-metal fast-breeder fission reactor by at least one order of magnitude. While this fact does not in itself demonstrate that fusion reactors are safer than fission reactors, it does indicate that the technical problems of ensuring fusion safety probably will be simpler. Both types of reactor require stringent design constraints in this respect.

The problems of an accidental or forced shutdown involve two major factors: (1) the afterheat associated with loss of coolant of the plasma-facing structure and in the lithium blanket and its structural elements, and (2) the energy storage of the magnets in the confinement system. The afterheat depends markedly on the structural materials (silicon carbide, vanadium, and ferrite steels are the three current candidates) associated with the blanket. The duration of the cooling period may vary from hours to years. These materials have biological-hazard potentials lower than plutonium, but still they are not negligible.

The forces associated with the large magnet structures at the high fields required for fusion are of large proportions. During normal operation, for example, in a typical tokamak reactor, the force tending to push the toroidal field coils toward the center is 10^8 lb per coil at a field of 150 kG.

The stored energy in the system can be as high as 10^{11} J. Uncontrolled quenching of the field can produce extremely high voltages, so the insulators associated with the magnetic system must have a high safety factor. Even in normal operation, when the field energy is purposely dissipated by passage through a resistance bank, a residual amount of the energy, of the order of 1%, may remain. This residue (10^9 J of energy) is still enormous by conventional standards.

The foregoing examples illustrate the nature and extent of the engineering problems that will remain to be solved after controlled fusion itself is demonstrated and reduced to practice.

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5.5 INDUSTRIAL COGENERATION

5.5.1 Cogeneration Defined

Cogeneration is a highly efficient means of generating heat and electric power at the same time from the same energy source. Displacing fossil fuel combustion with heat that would normally be wasted in the process of power generation, it reaches efficiencies that can triple, or even quadruple, conventional power generation. Although cogeneration has been in use for nearly a century, in the mid-1980s relatively low natural gas prices made it a widely attractive alternative for new power generation. In fact, gas-fired cogeneration is largely responsible for the decline in conventional power plant construction that occurred in North America during the 1980s. Cogeneration accounted for a large proportion of all new power plant capacity built in North America during much of the period in the late 1980s and early 1990s.

Cogeneration equipment can be fired by fuels other than natural gas. There are installations in operation that use wood, agricultural waste, peat moss, and a wide variety of other fuels, depending on local availability.

5.5.2 Siting Cogeneration Plants

Because it is impractical to transport heat over any distance, cogeneration equipment must be located physically close to its heat user. Cogeneration plants tend to be built smaller, and owned and operated by smaller and more localized companies than simple-cycle power plants. As a general rule, they are also built closer to populated areas, which cause them to be held to higher environmental standards.

In northern Europe, and increasingly in North America, cogeneration is at the heart of district heating and cooling systems. District heating combined with cogeneration has the potential to reduce human greenhouse gas emissions by more than any other technology except public transit.

5.5.3 Basic Concept of Cogeneration

Conventional power generation is based on burning a fuel to produce steam. It is the pressure of the steam that actually turns the turbines and generates power. This is an inherently inefficient process. Because of a basic principle of physics, no more than one-third of the energy of the original fuel can be converted to steam that generates electricity.

Cogeneration, in contrast, makes use of the excess heat, usually in the form of relatively low-temperature steam exhausted from the power generation turbines. Such steam is suitable for a wide range of heating applications, and effectively displaces the combustion of carbon-based fuels, with all their environmental implications.

In addition to cogeneration, there are a number of related technologies which make use of exhaust steam at successively lower temperatures and pressures. These are collectively known as combined-cycle systems. They are more efficient than conventional power generation, but not as efficient as cogeneration, which normally produces about 30% power and 70% heat. Combined cycle technologies can be financially attractive despite their lower efficiencies, because they can produce proportionately more power and less heat.

Cogeneration equipment recaptures the exhaust and water heat of power generation equipment and converts it into hot water, steam, space heat, process heat, air conditioning, and many other useful purposes. No additional fuel is used to provide this source of “free” energy. Figure 5-65 illustrates the process by which this “free” energy is captured and utilized.

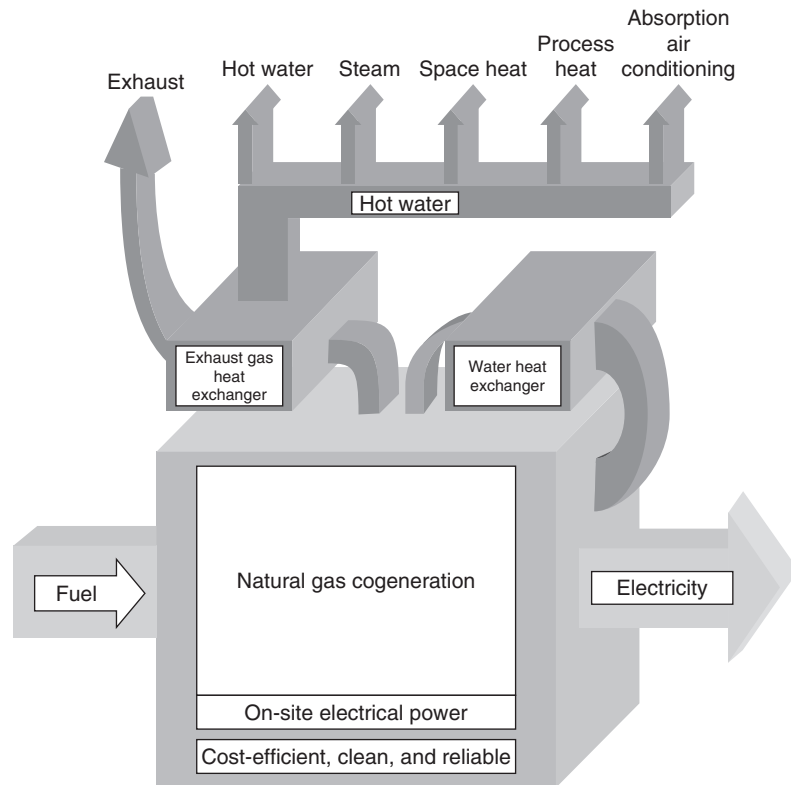


FIGURE 5-65 Diagram of the cogeneration process.

5.5.4 Advantages of Cogeneration

Cogeneration technology provides greater conversion efficiencies than traditional generation methods since it captures heat that would otherwise be wasted. This can result in more than doubling the thermal efficiency.

By recycling the waste heat, cogeneration systems achieve electrical efficiencies of 50% to 70%; a dramatic improvement over the average 33% efficiency of conventional fossil-fueled power plants. Higher efficiencies reduce air emissions and leading greenhouse gases, which are associated with climate change.

Environmental Issues. Carbon dioxide emissions will also be substantially reduced. Cogeneration systems predominantly use natural gas, a fuel source which emits less than half the greenhouse gas per unit of energy produced than the cleanest available thermal power station.

It is claimed by the Australian Cogeneration Association that cogeneration systems are more environmentally friendly, flexible, efficient, and can be more cost effective than traditional systems—especially when network costs and losses are taken into account.

Trigeneration provides even greater efficiency than cogeneration. Trigeneration is the conversion of a single fuel source into three useful energy products: electricity, steam or hot water, and chilled water. Trigeneration converts and distributes up to 90% of the energy contained in the fuel burned in a turbine or engine into usable energy. This is excellent efficiency compared to the 30% conversion that is typical for the standard electric generator.

5.5.5 Where Is Cogeneration Being Used?

A 1996 survey conducted by the International Cogeneration Alliance showed that the Netherlands was the world's leader of cogeneration use closely followed by two other Scandinavian countries—Denmark and Finland.

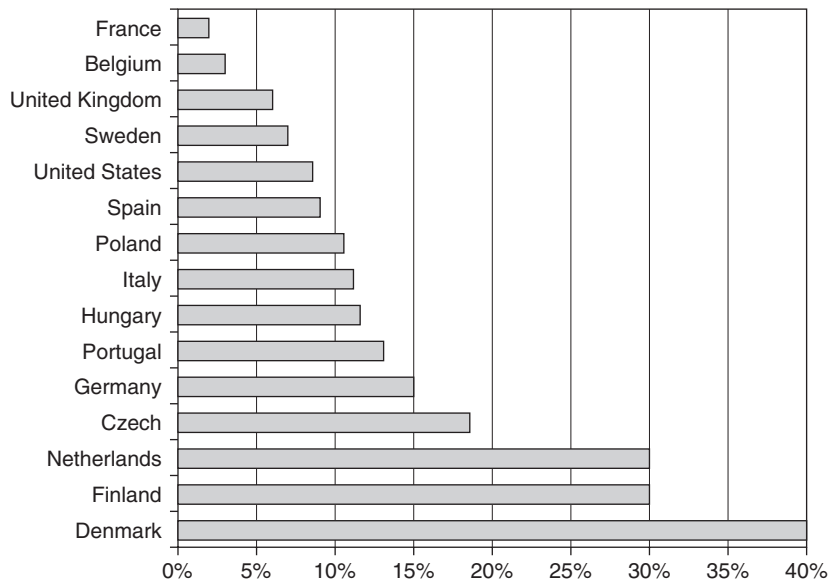


FIGURE 5-66 Percentage of national power production generated by cogeneration systems. *Source:* American Council for an Energy-Efficient Economy.

In 1900, half the electricity generated in the U.S. came from plants that also provided industrial steam or district heating. By the 1970s, cogeneration had fallen to less than 5%, but interest in this technology is being renewed.

The U.S. Department of Energy (DOE) set an initiative to double the use of combined heat and power systems in the United States by 2010. The U.S. National Energy Policy recommended that the president direct the U.S. Environmental Protection

Agency (EPA) to take an active role in promoting cogeneration. While progress is slow, some states, along with EPA and DOE, are trying to lower regulatory hurdles for combined heat and power systems.

Figure 5-66 shows the percentage of national power production generated by cogeneration systems in 1997 in a variety of European countries, along with the United States.

As older power plants age and need to be replaced, and as competitive pressures to cut costs and reduce emissions of air pollutants and greenhouse gases increase, owners and operators of industrial and commercial facilities will be actively looking for ways to use energy more efficiently.

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